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Overview of JET Results

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ABSTRACT
In the past 2 years, JET has contributed with a wealth of results to nearly all topics of the science of ITER.Significant developments in the reference operational scenarii of ITER has been obtained. Scaling studies of various physical properties of ELMy H-Mode and Advanced Scenarii have been refined. Progress in real-time control techniques, which have greatly contributed to the successful developments in ELMy H-Modes and Advanced Scenarios, is illustrated. Detailed studies have been undertaken related to the first wall, with much more details on the nature of the ELMs, the impact of disruptions, tritium retention and erosion and co-deposition. Major advances have been made in heating systems, with a proof of principle test of the conjugate T ICRH scheme and demonstration of coupling of LH waves at large plasma-launcher distances. Significant progress has been made in various diagnostics at JET, more in particular in Burning Plasma Diagnostics.

INTRODUCTION
JET's technical capabilities (major radius of 3m, plasma current up to 5MA in the present divertor configuration, ITER-relevant plasma triangularities, tritium operation and use of beryllium) allow plasma physics studies in conditions very close to those of a burning plasma. JET is therefore in a unique position to provide key contributions in preparation of ITER operation, in particular: (1) the ELMy H-Mode and Advanced Tokamak regimes foreseen to be used on ITER are further developed and their scaling as well as underlying physics studied in an as relevant as possible (ρ^*, ν^*, β_N) range; (2) significant progress is madeon burning plasma physics; (3) ITER-relevant diagnostic and heating techniques are being tested.

During the period 2003-2004, the use of JET was very intensive, with 165 days of experimentation. Of these, a total of 115 days were devoted to pure D plasmas, 10 days for experiments with reversed plasma current and toroidal field in D plasmas, 20 days for studies in H and He plasmas. and 20 days for a dedicated campaign with the use of tritium (Trace Tritium Campaign - TTE) conducted for the third time in the history of JET in October 2003 and mainly devoted to transport and particle confinement studies. The removal of the septum in the divertor (Fig.1) provided additional flexibility in plasma configurations. About one third of the experiments were conducted in the framework of the International Tokamak Physics Activity (ITPA),and several results presented below have gained from these collaborations. Overall, scientific progress on JET has profited largely from technological developments in control, heating and diagnostic systems.

Progress in preparation of ITER operating plasma scenarios is summarised in Sections 2 (performance), 3 (physics understanding) and 4 (real time control). Burning plasma physics results are presented in Section 5. The large size and plasma current of JET allow the effects of large disruptions and ELMs to be studied; in particular key issues in preparation of ITER operation are addressed, such as power deposition, erosion, sputtering and forces on first wall and divertor elements, with results reported in Section 6. Progress reached in the physics of ICRF heating and in ICRF and LH coupling are reported in Section 7. The increasingly demanding experimental conditions and requirements on the quality of measurements constitute a strong driver for diagnostics development, with progress presented in section 8, particularly emphasizing burning plasma diagnostics. We limit ourselves in this paper to a concise selection of the results obtained in the TTE Campaign (See Sections 3, 5, 7 and 8) since a detailed overview is presented in another paper to this conference (REF D.STORK). More details on the topics covered in this paper can be found in [Loarte,Sartori, Saibene, Tuccillo, ... all IAEA contributions of JET].

2. PERFORMANCE OF ITER OPERATING SCENARIOS

The ITER baseline scenario which is presently foreseen for reaching a power amplification Q=10 is the ELMy H-mode. Consolidation of this operational scenarios has made continued progress, aiming atrefining the fusion performance prediction and finding ways to reach higher fusion performance on ITER. Results are presented in section 2.1 and in section 3. Obtaining tolerable ELMs in this scenario remains a high priority topic. ELMs and other phenomena are discussed in Section 6.

The benefit of having steady state (or near-steady state) operations in a future tokamak fusion reactor has also triggered efforts toward the development of the so called Advanced Tokamak (AT) regimes, which would allow for a large fraction of non inductively driven current. Advanced Tokamak (AT) regimes discussed in this paper cover the purely non-inductive ‘steady-state’ mode of operation, characterized by areversed q profile (q_0=2-3, q_{min}=1.5-2.5), and by the presence of an Internal Transport Barrier (ITB) and the ‘hybrid’ regime (q_0>~1, monotonic q profile, also called improved H-mode), which is an intermediate scenario between the inductive H-mode and the fully non-inductive current drive regime operating scenario. The results on ITB plasmas are mainly reported in sections 2.2, 3 and 4, and those related to the hybrid mode in section 2.3 and 3.

2.1. ELMY H-MODE

High density high confinement operation. The ELMy H-Mode is the reference operating scenario for ITER, projected to deliver a fusion output equivalent to Q=10. Although the realization of this scenario atthe ITER working point has been demonstrated in the last years on JET, optimization continues, in particular in boosting further the performance in confinement and density, and in documenting better the scaling ofenergy and particle confinement and the threshold power for the L-H transition.

ELMy H-mode plasmas at ITER-like triangularities (δ_{av}=0.47) have been obtained at $H_{98(y,2)}$=1, n/n_{GW}>0.85 for plasma currents up to 2.5MA corresponding to q_{95}=3.6 and at somewhat lower δ_{av}=0.42, q_{95}=3.0 up to 3.5MA. Large NTMs were successfully avoided by implementing operational strategies, which prevent large sawteeth or first ELMs, both of which can trigger NTMs via large seed islands. It has been possible to increase the density further to $n/n_{GW} \approx 1.1$ albeit with a slight reduction in confinement $H_{98(y,2)}$~0.9. Under these conditions, the H-mode pedestal enters the mixed Type I/II ELM regime. With increasing density, this regime is characterised by an increasing pedestal pressure caused by an increased pedestal density at roughly constant temperature, a decreasing Type I ELM frequency and increased inter ELM losses accompanied by enhanced low frequency magnetic fluctuations.

L-H Transition. Experiments were carried out on JET to study the effects of divertor geometry and plasma shaping on the power needed for the L-H transition, $P_{L\text{-}H}$. Equivalent septum configurations with theSeptum Replacement Plate divertor (MkII-SRP), previously run in the Gas Box Divertor (MkII-GB) have shown that the presence of the septum lowers $P_{L\text{-}H}$ by 20%. This result can largely be explained by aconstant pedestal T_e at the L-H transition both with and without the septum and the differences in power needed to reach it. Additional experiments show that with a lowered X-point height, plasmas run with the MkII-SRP divertor reproduce a significant decrease in PL-H (up to a factor of two for a variation in X-point height of about 16cm) and pedestal electron temperature T_e, first observed with the MkII GB septum divertor. The experimental study of plasma shaping effects have shown δ^{up} to have no influence on $P_{L\text{-}H}$ orpedestal T_e/T_i. Attempts to study the influence of δ_{low} are so far inconclusive, as such variations require avariation in the X-point height. Finally, a comparison of the reversed B_t discharges with reference plasmas with the standard B_t direction shows no significant difference on $P_{L\text{-}H}$ in JET, in contrast to ASDEX-Upgrade [Suttrop PPCF 39 2051-2066(1997), Ryter PPCF 40 725-729 (1998)] and DIII-D [Carlstrom PPCF 40, 669-672 (1998)].

Deuterium only fuelling scans at q_{95}=3.4 have confirmed the Borrass-Lingertat-Schneider scaling (K.Borrass, NF 44, 752-760 (2004)) of the H to L back-transition density, irrespective of triangularity, suggesting that the intended Greenwald fraction in ITER would be inaccessible by gas injection alone. This could partially be offset by density peaking, since studies of Type III H-modes with mid-radius collisionality, brought within a factor of two of the ITER value, have indicated that the particle diffusivity was about 5 times lower than assumed in current modelling, with consequently moderate profile peaking. [#see paper by H Weisen]

2.2. ADVANCED MODES WITH INTERNAL TRANSPORT BARRIERS

The closeness of the JET scenarios to those foreseen for ITER in terms of physics parameters (simultaneously ρ^*, ν^* and edge conditions) allows detailed investigations of possible solutions to e.g. integrated Real Time Control (RTC) of AT regimes, their compatibility with low momentum input and ITB resilience to ELMs. During the 2003-2004 experimental Campaigns, the Advanced Tokamak studies on JET progressed significantly thanks to: (i) an enhanced RTC system, now allowing to control a multitude ofsignals on-line, with as highlight the simultaneous control of the pressure and q profile (See Section 4) [Moreau this conf.], (ii) further progress in Lower Hybrid (LH) coupling (See Section 7) [Mailloux this conf.], (iii) the increased plasma shaping capability deriving from removal of the septum from the divertor (see figure $$$) and (iv) a further increase in the neutral beam heating power.

While the inductive H-mode is relatively well explored, an open issue is how the presently developed non-inductive current drive regimes with improved core confinement will extrapolate to next-step experiments. During the 2003-2004 experimental campaigns, the physics of improved core confinement regimes and the conditions for the formation and sustainement of AT regimes [Tuccillo, IAEA] have been further investigated. An extensive database [Litaudon, PPFC 46, 2004 A19-A34] on the advanced mode of operation has been assembled covering a wide range of plasma parameters (q-profile, T_i/T_e, R/LT gradient length, triangularity, Larmor radius ∇*, collisionality ∇*e, Mach number M_∇ ...).

High density operation. Until recently, one of the main difficulties was the low target density (n/n_{GW}<0.5) of AT discharges. ITBs on JET can now be triggered with core density close to or above the Greenwald density using an adequate timing of LH and pellet injection to reach a reversed q profile and peaked density profiles. Under these conditions, ITBs have been observed on the profiles of ion- and electron temperature, density and toroidal rotation and at a bulk plasma rotation 4 times lower than ITB plasmas atlow target density and high NBI input power. This has been obtained with simultaneous NBI (8-9MW) and ICRH (6-7MW) ($$#57941) heating leading to ITBs with T_e~T_i~7keV. The foot of the ITB in this case is typically inside R=3.5m, i.e. inside mid-radius.

Large Internal Transport Barriers. Large ITBs (R>3.5m) can be formed by reducing the magnetic shear close to a low order rational surface in the vicinity of the plasma boundary. This has been done in two different ways: (i) in the current ramp-up at high plasma current (I_p /Bt~3.0MA/3.2T) and low q_{95} (~3) relying on the slow penetration of the off-axis ohmic current. At these high currents only low triangularity discharges could be explored up to now on JET (lower disruptive forces); (ii) at low plasma current (I_p<=2MA) and high q_{95} (>~5) in the presence of a large fraction of off-axis driven current (LHCD andbootstrap current). The latter have been realised in both high and low triangularity discharges. At low triangularity, the ITBs are located at R=3.6m, close to q=3 surface, generally do not accumulate impurities (due to the weaker density and temperature gradients at the obtained high radii of the ITB) and can coexist with other more internal barriers (ITB located at R~3.3m), in a zone with strong negative shear. These discharges show type-III ELMs and are reproducibly sustained for a time close to neo-classical resistive time(~10s).

The largest ITBs (R~3.7m) have been found at high triangularity ∇~0.45 in 3.4T/1.5MA at q_{95}~7.5 plasmas with combined NBI= 18.5MW, ICRH = 2.5MW and LH=2MW (Pulse No: 62293) heating. (Rimini this conference). At large triangularity the main difficulty is to optimize simultaneously the edge conditions (avoiding type I ELMs) and q-profile to maintain the barrier. Despite the closeness of foot of the barrier to the edge pressure pedestal, type I ELMs can be moderated using Ne seeding. (Fig.2). The performance reached with this scenario is as follows: ∇_p=1.5, H_{89}=1~1.8, ∇_N~1.8, n/n_{GW} ~70%, P_{rad}/P_{tot} ~50% andV_L~0.05.

Long Pulse Operation. Highly non-inductive reversed shear operations have been achieved in 3T/1.8 MA discharges where a combination of 10MW NBI, 5MW ICRH and 3MW LH power is applied. These pulses have been optimised at reduced NBI power such that the power delivered by each beam box could be applied sequentially in time. At low beam power, the peak performance is reduced to H_{89}=2,∇_N~1.66 and with a bootstrap fraction of about 33%. In these conditions the pulse duration has been extended to the maximum duration allowed by the JET subsystems (20s ~ 2_t res) with a record of injected energyE~326 MJ.

ITB formation. In order to investigate the physics and extrapolation of the ITB formation in terms of non-dimensional parameters identity experiments between JET and ASDEX-Upgrade have been undertaken at similar q-profile (i.e same q_0~2, q_{95} and shape), ∇*, ∇*. A similar phenomenology of the ITB formationand erosion due to Type I ELMs has been reproduced on JET. Moreover on JET, the ITB regime, transient on AUG, has been sustained on longer time scales (10∇ ∇_E) with barriers present both on ion- and electron temperature profiles through the mitigation of ELMs by Ne injection.

2.3. HYBRID MODES

Identity experiments have been also performed between JET and ASDEX-Upgrade to confirm the existence of the hybrid regime closer to ITER parameters (Joffrin, IAEA) (i.e. with monotonic q-profile and q_0>~1, q_{95}~4). JET experiments first focused on reproducing the hybrid regime with a careful match of the magnetic configuration, q profile, ∇* and plasma ∇ with those of ASDEX Upgrade hybrid plasmas. The performance of hybrid scenario plasmas on JET has been successfully verified up to ∇_N=2.8 at low toroidal field (1.7T), at both high and low plasma triangularity (0.2 and 0.45) and ∇* corresponding to the ASDEX Upgrade discharges. Stationary conditions have been achieved with a figure of merit for fusion gain $H_{89}.\beta_N/q_{95}^2$ reaching 0.42 at q_{95}=3.9 at a Greenwald density fraction of 0.7, equivalent with ∇*e ~ 0.08 or 4∇∇*$e,_{ITER}$.

Dedicated similarity experiments have then extended the hybrid regime operations on JET to lower ∇* (as low as ∇*=3.310^{-3} or 2∇∇*$_{ITER}$ and ∇*$_e$ ~ 0.06 or 3∇∇*$_{e,\ ITER}$) by increasing the toroidal field (B_t=2.4T and 3.1T). The operation at high current and field (i.e. ∇* closer to ITER and reasonably low ∇*) has yet been limited to lower ∇_N values due to limited heating power. The parametric range needs to be extended to values closer to the ITER values to allow a full set of scaling studies.

This regime has also been reproduced on JET at low plasma rotation with dominant RF heating (NBI=7.8, LH=1.2, ICRH=10MW) in plasmas with I_p/B_t= 2.6MA/ 3.2T, q_{95}=3.8 (Pulse No: 62789), resulting in q_o~1,T_i close to T_e, H_{89}=2, _N=1.55 with Type III ELMs (Fig. 3) [Gormezano, EPS]. Up to now no impurity accumulation has been seen in the hybrid regime.

The special importance of the hybrid mode is the avoidance of mechanisms of triggering deleterious NTMs by sawteeth (q_0>1). This provides the possibility of extending this mode towards high ∇_N operation. In JET, the hybrid regime has been operated with small NTM islands (2-3cm) up to high value of ∇_N (~2.8) reaching more than 90% of the ideal kink limit. In these discharges the ∇_N values approach the no wall limit defined empirically by ∇_N ~ 4l_i. Operation above this limit will require higher power, and this is part of future experimental campaigns on JET.

3. PHYSICS STUDIES RELATED TO FUSION PERFORMANCE OF ITER-RELEVANT PLASMA SCENARIOS

3.1. H-MODE CONFINEMENT SCALING

The scaling of energy and particle transport for ELMy H-Modes, in terms of non-dimensional parameters, has been studied with a series of ∇*, ∇ and ∇* scans. Tritium particle confinement was studied with trace tritium (n_T/n_D<5%) puffing and beam blips into a low q_{95}=2.8, low triangularity ∇=0.2 D plasma,energy confinement was studied in pure D plasmas.

Two ∇* scans for energy confinement were performed in the same scenario with both Type I (_t=2.6%, Ip=2-4MA) and Type III (∇_t=1.3%, Ip=1.3-4.3MA) ELMy H-mode plasmas. A gyro-Bohm like scaling was found for both the Type I ($\nabla_{ci}\nabla_E\nabla\nabla*^{3.2\pm0.4}$) and Type III ($\nabla_{ci}\nabla_E\nabla\nabla*^{-2.9\pm0.5}$) scans, consistent with that seen in IPB98 (y,2). A three point _* scan for tritium particle confinement showed a similar gyro-Bohm like dependence ($D_T/B_0\nabla\nabla*^{3.2\pm0.6}$)(0<r/a<0.45), but with a weaker Bohm-like dependence in the outer region ($D_T/B_0\nabla\nabla*^{1.9\pm0.4}$) (0.65<r/a<0.85).

Two two-point ∇ scans for energy confinement, were also performed in the same scenario, along with a three-point ∇ scan in higher q (q_{95}=3.2) and triangularity (∇=0.3) plasmas. In contrast to the IPB98 (y,2) scaling, a very weak beta dependence ($\nabla_{ci}\nabla_E\ \nabla\nabla^{-b}$, with b=-0.03 to +0.04) was found for energy confinement in the three scans, confirmed in complementary experiments on DIII-D [Petty C. C., *et al.*, Phys. Plasmas **11** (2004) 2514], consistent with transport dominated by electrostatic processes and in agreement with previous dedicated scans [Christiansen J. P. and Cordey J. G., Nuc. Fus. **38** (1998) 1757;Petty C. C., et al., Nuc. Fus. 38 (1998) 1183]. The two point ∇ scan for tritium particle confinement, by contrast, showed a positive impact of ∇ on particle confinement ($D_T/B_0\nabla\nabla^b$, with b=-0.34 to –0.55) which contradicts IPB98 (y,2). A model, based on neoclassical orbits in stochastic electromagnetic fields (Voitsekhovitch I., *et al., 'Trace Tritium transport in H-mode JET plasma with different density'*, In preparation for Phys. Plasmas), has been shown to reproduce the observed beta dependence of particle confinement, but a unified model, describing the ∇ dependence of particle and energy transport, remains outstanding.

A ∇* scan in a q_{95}=4.4, ∇=0.4 scenario, showed energy confinement having a negative dependence on ∇* ($\nabla_{ci}\nabla_E\nabla\nabla*^{-0.35\pm0.04}$), in contrast to the negligible ($\nabla_{ci}\nabla_E\nabla\nabla*^{-0.01}$) IPB98 (y,2) scaling. These new findings support electrostatic energy confinement scalings, which predict a moderate increase in energy confinement ($\nabla\nabla_E/\nabla_E$=2-28%, depending on the details of the scalings used) for the ITER n/n_{GW}=0.85, ∇_N=1.8 baseline, but a dramatic increase at higher ∇_N ($\nabla\nabla_E/\nabla_E$ =23-50% for ∇_N=3.0), which would allow operation with fusion gains of Q>15 at n/n_{GW}=0.85 and even higher values, Q>20, for $n/n_{GW}\nabla$1.0. The impact of the unfavourable increase of particle confinement, observed in the JET trace tritium experiments, on ITER operation at high ∇_N remains to be assessed. [#see paper by D.C McDonald et al., this conference]

3.2. PARTICLE TRANSPORT EXPERIMENTS DURING THE TRACE TRITIUM CAMPAIGN

The JET Trace Tritium Experimental (TTE) campaign used deuterium plasmas with trace amountsof tritium (T) ($n_T/n_D<3\%$) to investigate thermal fuel-ion transport, fast particle dynamics and heating andcurrent drive physics. The non-dimensional parameters ∇∇, ∇∇, ∇ and q_{95} were varied in pairs of sawtoothing ELMy H-Mode discharges, to compare the scaling for the transport coefficients of fuel particles. The main results can be summarized as follows. Tritium particle transport is generally above neoclassical levels (as calculated by NCLASS [Houlberg, Phys of Plasmas, 4 (1997) 3230]) and only under two circumstances areneo-classical values approached: at high density in high current, low q_{95} discharges [Belo EPS 2004] and inthe transport barrier region of ITB discharges [Mailloux EPS 2004]. In hybrid scenarios ∇_{pT}* improved by~50% in triangularity scans (∇ = 0.2 - 0.46) at constant energy confinement. Comparing different regimes (ELMy H-mode, ITB plasma, and Hybrid scenarios) outside the central plasma region (0.65<r/a<0.85), the tritium diffusion coefficient (D_T/B_∇) scaling is close to Gyro-Bohm ($\sim\nabla_\nabla^{\nabla\text{-}3}$, where $\nabla_\nabla^{\nabla}=q\nabla^{\nabla}$), but with anadded inverse ∇ dependence, in contrast to energy transport which shows no dependence on ∇ as reported in section 3.1 [McDonald et al, IAEA 2004].

3.3. TRANSPORT PHYSICS AND TESTS OF THEORETICAL MODELS

Theory predicts that both thermo-diffusion and magnetic field curvature contribute to the turbulent pinch [X.Garbet et al., PRL, 91, 035001/1-4]. Also curvature pinch depends on magnetic shear and the peaking factor decreases with collisionality. These predictions have been tested against experimental resultsat JET. It has been verified that the pinch velocity increases with magnetic shear in L-mode plasmas [*L. Garzotti et al., Nucl. Fusion* ***43****, 1829 (2003), H. Weisen et al., Plasma Physics and Control. Fusion* ***46****(2004) 751*]. In H-mode, the density peaking is sensitive to collisionality, consistently with ASDEX-Upgraderesults. It is found that at high collisionality, the pinch velocity is close to the Ware value, whereas it is larger at low collisionality [*H. Weisen et al., submitted to Plasma Physics and Control. Fusion, M. Valovic et al., toappear in Plasma Physics and Control. Fusion*]. This suggests that density profiles in ITER may be more peaked than presently assumed. RF heating is found to flatten density profiles. When collisionality is large, this is interpreted as an increase of the turbulent diffusion coefficient driven by ion heating while the pinchvelocity stays close to the neoclassical value. For low density plasmas, the interpretation rather relies on electron heating inducing the weakening (and may be reversal) of pinch velocity predicted by theory when the turbulence moves from ion to electron dominant instabilities. For these reasons, it is believed that alpha particle electron heating in ITER (where ion instabilities will still be dominant) will not have a dramatic flattening effect on density profiles. RF heating was also found beneficial to control impurity accumulation via a decrease of impurity inward convection [*M. E. Puiatti, Plasma Physics and Control. Fusion* ***44*** *(2002)1863*], although it is still not clear if the effect is due to a reduction of the neoclassical impurity pinch associated to density flattening or to a modification of the significant turbulence driven component of impurity convection.

Heat modulation experiments have been undertaken at JET using ICRH in mode conversion scheme. These experiments have been analysed using a critical gradient transport model, stability analysis and predictive modelling. Instability thresholds are found to be in the expected range for micro-instabilities intokamaks. Electron stiffness was found to cover an unexpectedly wide range of variation [*X. Garbet et al.,Plasma Phys. Control. Fusion* ***46****, 1351 (2004)*]. However the Weiland transport model was able to reproduce experimental results [*P. Mantica et al., in Proceedings of the 31st EPS Conference on PlasmaPhysics, London, 2004*]. A correlation was found between electron stiffness and the ion temperature gradient length [*P. Mantica et al., this conference*]. This observation suggests that some interplay exists between electron and ion heat channels. The applicability of a critical gradient model in the case where ion and electron modes are linearly unstable will thus have to be further investigated. No firm conclusion can yet bedrawn regarding profile stiffness in ITER.

3.4. RESISTIVE WALL MODES AND HIGH BETA OPERATION OF ADVANCED MODES

In advanced tokamak operating scenarios, such as those foreseen for ITER and compatible with the steady-state operation of a power plant, the ultimate performance limit is set by resistive wall modes (RWMs) [Liu Y Q et al, Nuc Fus **44** (2004) 232]. Therefore, the RWM must be stabilized in order to achievesteady state operation with high plasma pressures. The RWM is a kink mode whose growth rate is largely governed by the tokamak wall time but whose stability is related to damping depending on the velocity ofplasma rotation. Calculations using MARS-F code [Liu Y.Q., et al., Phys. Plasmas **7** (2000) 3681] show that the critical rotation required for RWM stabilisation depends sensitively on the damping models.

The damping of stable RWMs may be determined experimentally by measuring the response to n=1helical magnetic perturbations from coils external to the plasma - in JET, saddle coil systems both internaland external to the vacuum vessel have been used for such studies [T C Hender et al, *Resistive Wall Mode Studies in JET*, this conference], - and then compared with the MARS-F predictions. The resonant field amplification (RFA) of this externally applied field has been measured for both DC and AC appliedmagnetic perturbations. RFA was observed in JET as ∇ increases, particularly beyond the no-wall limit, and comparison with MARS-F modelling shows that both a strong parallel damping or a semi-kinetic model [seeLiu] agree best with experiment. The occurrence of a critical flow velocity below which the RWM becomesunstable was found to be in agreement with modelling [Y Liu et al, *Feedback and Rotational Stabilization of RWMs in ITER*, this conference].

The results of these first European experiments on resistive wall modes in tokamak plasmas providea very important experimental validation of RWM damping models allowing for extrapolation to ITER, and it is found that the observed strong damping leads to a requirement for a flow of ~2 to 3% of the Alfvénvelocity at the plasma centre to stabilise the RWM. It is marginal whether the flow velocity in ITER willreach such values indicating that an active RWM control system will be a prudent option.

4. REAL TIME CONTROL OF ITER OPERATING SCENARIOS

Real-time control of the reference scenarios for ITER is of key importance. Two examples will illustrate this. For the ELMy H-Mode, where the plasma shape is a main element in determining the performance, high triangularity and elongation need to be maintained during the pulse, even in the presence of (large) variations of current or pressure profile. For Advanced Modes, the q-profile plays a crucial role in accessing high confinement, stability and current drive, and controlling its shape is therefore a key elementin establishing and maintaining high performance

4.1. CONTROL OF THE PLASMA SHAPE

A control system that is able to maintain the plasma shape in presence of large disturbances (e.ggiant ELMs and large variations of ∇_p and/or l_i) is a key element for performing successful experiments. TheeXtreme Shape Controller (XSC) system has been successfully installed and commissioned during a recent experimental Kampaign on JET. The description of the plasma shape is done with a set of geometrical descriptors (gaps), that define the distance along predefined directions of the last closed flux surface from the first wall. In order to control the overall plasma shape during a JET pulse, the XSC has been designed to implement a system in which all 8 poloidal coils are used as actuators for the real-time control of a large setof geometrical descriptors (48). Since the number of available actuators is only 8, an optimisation process that defines the sensitivity of each gap to a variation of the current of a given coil has been performed by means of a singular value decomposition (SVD) analysis. The design aims have been demonstrated during aset of high triangularity ITBs discharges in presence of quite large variations of ∇_p (∇_p up to 1.5) and/or l_i (l_i up to 0.5) [Sartori et al, SOFT].

4.2. CONTROL OF PLASMA PROFILES IN ADVANCED TOKAMAK REGIMES

To sustain ITBs on ITER time scales, a careful optimisation of plasma characteristics and power tailoring through integrated Real Time Control of ITB strength and profiles will be required. E.g. the presence and size of a ‘current hole’ in discharges with deep reversed shear should be carefully control ledduring the high D-T fusion yield phase of future experiments, since (i) these configurations are characterised by a low confinement of fusion born alphas as measured in JET where traces of Tritium have been injected and (ii) deleterious TAE modes could be triggered with large values of q_{95}. Past experiments [D. Mazon, PRL] focussed on the separate control of the maximum normalised electron temperature gradient ∇^*_{Te} (see definition in [$$]), on one hand [$$], or of the safety factor profile on the other hand [$$], in different discharges. A major challenge still remained to simultaneously control both the current and pressure profiles, which mix up the resistive and confinement time scales in a non-linear way. Multi-variable model-based techniques (Moreau, this conference) for the control of the current and/or the pressure have now been developed. The technique aims at minimising an integral square error signal which combines the two profiles, rather than attempting to control plasma parameters at some given radii with great precision. The resulting fuzziness of the control scheme allows the plasma to relax towards a physically accessible nonlinear state which may not be accurately known in advance, but is close enough to the requested one toprovide the required plasma performance. Closed loop experiments have allowed to satisfactorily reach different target q and ∇^*_{Te} profiles, and, to some degree, to displace the region of maximum electron temperature gradient. The control has also shown some robustness in front of rapid transient events like TypeI ELMs, and spontaneous emergence or collapses of the ITBs. An improvement of the proposed technique could consist in identifying a dynamical linear model which would allow to design a two-time-scalecontroller, perhaps more suited to control rapid plasma events while slowly converging towards a requested steady state.

4.3. CONTROL OF IMPURITY CONTENT, CONFINEMENT AND DENSITY

A dual feedback system was developed in order to control both the values of $H_{98(y,2)}$ and $\nabla=P_{rad}/P_{tot}$.The D_2 puffing rate and the Ar seeding rate are used as actuators. Confinement and radiation were chosen as controlled parameters since they were the best correlated quantities with the actuators. Furthermore this choice allows to maximise the D_2 fuelling rate for a given confinement and radiation. The feedback scheme uses a 2 by 2 control matrix, which is established from open-loop experiments with step requests for the actuators and is valid for a certain range around the chosen operational point, which results from the strong non-linear couplings between the various physical quantities to be controlled. The plasma model – essentially described by this matrix - is then tested on several discharges from the existing database. Once the plasma model is responding satisfactorily it is implemented in the feedback loop in order to test the stability of the scheme and optimise the Proportional, Integrated and Differential (PID) time constants of the loop. This procedure has been successfully applied to Ar seeded discharges with additional T puffing during the TTE experiments. The discharges obtained show simultaneously high confinement $H_{98(y,2)}=1$, $\nabla_N>1.8$ and high density $n/n_{GW}=1$for the whole feedback control phase (>~5s).

5. BURNING PLASMA PHYSICS [STORK ET AL., THIS CONFERENCE, KDZASTROW, EPS 2004]

5.1. CONFINEMENT OF ALPHA PARTICLES IN ELMY H-MODE AND CURRENT HOLE DISCHARGES

Alpha particle confinement in ELMy H-Mode and Current-Hole (CH) plasmas, was studied [KiptilyPRL, Sharapov, this conf.] by detecting ∇-rays from reactions between fusion alphas (generated by Tritium Neutral Beam blips, TNB) and beryllium impurities ($^{9}Be(\nabla,n\nabla)^{12}C$) in deuterium plasmas after TNB turn-off. ∇-ray emission decay times measured the fusion-∇ population slowing-down time (∇_∇s). In CH discharges ∇_∇sdetermined in this way was ~5 times lower than classical values, indicating ∇-confinement degradation, dueto orbit losses as predicted by 3-D Fokker Planck code. Studies in ELMy H-Mode plasmas confirmed the classical picture of the alpha particle slowing down.

5.2. FAST ION DYNAMICS

Fast ion confinement and instabilities (Toroidal Alfvén Eigenmodes (TAEs), Alfvén Cascades (ACs)) are key elements in determining the plasma heating, fast ion dynamics and transport in next-step burning plasma devices such as ITER. Several different groups of fast ions will be present in such a machine (^{4}He, D, ^{3}He and H resulting from fusion reactions, MeV range injected beam ions and ICRH-acceleratedions) and it is essential to measure and understand the consequences of transport and instabilities for each group of fast ions separately. These issues have been addressed in experiments on JET, with innovative diagnostic techniques, in conventional and shear-reversed plasmas, exploring a wide range of effects.

Using third harmonic ICRF acceleration of ^{4}He beam ions in ^{4}He plasmas fast alpha particle populations [refSharapov, ref to Mantsinen[1] Mantsinen M., Mayoral M.-L., Kiptily V., Sharapov S., Bickley A., de BaarM., Eriksson L.G., Hellsten T., Lawson K., Nguyen F., Noterdaeme J.M., Righi E., Tuccillo A., Zerbini M., and EFDA-JET Contr., ‘‘Alpha Particle Physics Studies on JET with ICRH-Accelerated He4 Beam Ions’’,*Energetic Particles in Magnetic Confinement Systems, (Gothenburg, 2001), Vol. OT-28, IAEA*] have beencreated in a ‘D-T neutron-free’ environment. Simultaneous measurements of spatial profiles for fast Deuterium and alpha particles were performed successfully for the first time with gamma-ray tomography (See Section 8), confirming the agreement of fast particle dynamics with the classical theory of fast ion orbits in plasmas with monotonic and strongly reversed magnetic shear. **FIND OUT WHAT ARE THE LIMITS (E.G. TF S1 CRITICISES THE RELEVANCE OF THSE STUDIES, LACK OF RELEVANT H-MODE ETC.; AND FIND OUT WHICH ARE THE PROSPECTS FOR PROGRESS)**

With ICRF minority heating of ^{3}He (in ^{4}He plasmas) and H (in D plasmas) tails of fast ^{3}He and H were created. Direct profile measurements of the created fast ions were also performed using gamma-ray detection (see Section 8). Time-resolved measurements of the fast ion profiles were performed with a time resolution up to 10msec (which is possible due to the high intensity of the gamma radiation for the reactionsconsidered), while the energy spectrum was measured with a gamma-ray spectrometer and a neutral particleanalyser. A strong coupling between Alfvén modes inside and outside the q=1 surface was observed in plasmas with a flat and monotonic q profile, and represents a significant loss channel for fast particles.

5.3. ALFVEN WAVE EXCITATION STUDIES

In preparation.

5.4. CONTROL OF FAST PARTICLE INDUCED SAWTEETH

Alpha particles are likely to stabilize sawteeth in machines with strong central fusion heating. Seedislands appearing after the crash of such sawteeth tend to destabilize NTMs leading to a strong degradation of the confinement time. Localized current drive near the q=1 surface can destabilize sawteeth and the smaller crashes of sawteeth with shorter sawtooth periods do not lead to those detrimental effects. To showclearly the capability of ICRF to destabilize large sawteeth stabilised by fast particles, large sawteeth werefirst created and then destabilized. ICRF power of 3 MW in the (H)D minority scenario with a concentration of n_H/n_D ~5% and central resonance using a co-current directed (inward pinch of fast particles) spectrumefficiently created long sawteeth with a period of up to 250ms in 2.6MA plasma discharges. A power of 3MW in counter-current propagating ICRF waves at the q=1 surface on the high field side decreased the sawtooth period to 80ms [Eriksson L.-G., Mueck A., Sauter O., Coda S., Mantsinen M., Mayoral M.L.,Westerhof E., Buttery R.J., McDonald D.C., Johnson T., Noterdaeme J.-M., de Vries P., and contributorsJ.E., ‘‘Destabilisation of Fast Ion Induced Long Sawteeth by Localised Current Drive in the JET Tokamak’’, Phys. Rev. Letters **92** (2004) 235004-1 to 4]. The method is sensitive to the correct positioning of the resonance layer to within 5cm.

6. DIVERTOR AND FIRST WALL STUDIES IN PREPARATION OF ITER

6.1. EROSION AND CO-DEPOSITION AT FIRST WALL COMPONENTS.

Material erosion, its subsequent transport in the plasma and final co-deposition with the plasma fuelis an important issue facing operation with the longer pulses of next generation devices. An idea of the scale of the problem can be seen by considering that a single ITER pulse is equivalent to half a JET year of operation in energy input and ~5 JET years of divertor ion fluence.

Recent analysis through a variety of independent approaches has shown remarkable agreement between estimated rates of carbon impurity evolution from main chamber sources and inner divertor deposition. This provides strong support for an emerging view in the tokamak community that erosion occurs predominantly on main chamber surfaces (via charge-exchange neutral or ion impact) and that the resulting impurities are convected preferentially to the inner divertor during operation with the normal toroidal field direction (ion B x ∇B drift downwards). Such preferential deposition has also been directly confirmed on JET using dedicated $C^{13}H^4$ gas injection studies [54] and through the use, unique to JET of beryllium evaporation in the main chamber. Strong parallel flows with Mach numbers between 0.2 and 0.6 have been measured near the top of the machine, directed from outer to inner divertor for normal field and are a clearc andidate for the source of convective, long-range material transport. Considerable simulation efforts using both the EDGE2D-Nimbus [55] and B2.5-Eirene code [56] packages are underway, though the measured flows remain anomalously high compared with numerical predictions when classical drift terms are included. A recent unexpected and unique finding has has been that the outer divertor, normally a region of net erosionin JET, becomes a zone of net deposition following toroidal field reversal [57]. This is accompanied by the observation of stagnant SOL flow in the main SOL and has added further importance to the code simulation efforts including drift physics. Whenever carbon is present (as in JET), however, the situation is complicated by plasma chemistry and a complete description of the erosion-redeposition process and subsequent short-range carbon transport must also be included [V. Philipps et al., this conference].

6.2. TRITIUM RETENTION

Ongoing ex-situ analysis of graphite tiles from a whole series of previous JET divertors is revealing average fuel retention rates of ~3%. For example, during the MKIIGB divertor phase, about 766g of deuterium fuel was injected and 22g found in the divertor following post-mortem analysis. This should be compared with the results of tritium balance during the JET DTE2 campaign in which 10.5% retention was obtained after an extensive clean-up campaign. Recent in-situ measurements of material accumulation in remote areas indicates that this overall reduction in retention may largely be due to regular operation in recent years with vertical target configurations in the gas box divertors compared with the earlier horizontal target operation, implying an important role for divertor geometry in determining the extent of migration and retention [Philipps et al., this conference]. A very recent gas balance experiment in JET [Loarer et al., this conference] in which ~40 reproducible, H-mode discharges with high fuelling were executed over a full dayof operation showed essentially zero long term retention within measurement accuracy..

6.3. DISRUPTION STUDIES

Considerable effort has been devoted on JET towards improving both the database and understanding of energy and particle transients and the resulting transport to first wall surfaces. Analysis of energy flow from the bulk plasma just before and during disruptions across a wide range of disruption types [50] demonstrates that most discharges reach disruption with a small fraction of the full performance thermal energy. The exceptions are disruptions following VDEs and ITB collapses (which also have the shortest thermal quenches). This reduced thermal load considerably improves life expectancy of the ITER divertor and plasma-facing components. In addition, energy deposition has been observed outside the divertor [51], potentially improving the scope for ITER divertor material options but with a possible impact, that requires further experimental clarification, on the current choice of a beryllium first wall. In fast current quench disruptions, the electromechanical load due to the induced currents represents one of the most severe designconditions for in-vessel components. Based on data from JET [52] and most other tokamaks [53], the minimum linear decay time normalised to the plasma cross-section extrapolates to a 40 ms disruption forITER. Contrary to expectations, the quench rate of high and low thermal energy disruptions does not appear to vary substantially.

6.4. EDGE LOCALISED MODES

Assuming that large Type I ELMs will have to be tolerated, and extrapolating current ELM data to ITER, expected divertor target erosion rates are still too high, although the improved JET database now estimates the expected ELM size to be only about a factor of 2 above the limit for target ablation (C or W targets). This is to be compared with earlier, pre-2002 predictions, that put the ELM deposited energy afactor of 5 above tolerable limits. In many cases, Type I ELMs on JET are also associated with rapid displacements of the divertor strike points, often over several cm. This is of potential importance for ITER with regard to power deposition. Very recent measurements in the far SOL of JET [Pitts et al., to besubmitted - PRL] provide strong evidence that ELM ion energies remain high near the wall radius. This has implications both for the energy density on the first wall during ELMs but also for the degree to which ELMs may be tolerated in a reactor with a high Z wall. It has been found, moreover, that although ELM energy deposition becomes more balanced between inner and outer divertor targets at high I_p, larger ELM stend to deposit increasingly lower fractions (as little as 50%) of the pedestal energy in the divertor region [Eich et al., to be published in J.Nucl Mater (invited review paper at 16th PSI conference, Portland, Maine, USA, 22-26 May 2004], the rest being intercepted by main chamber wall surfaces.

High I_p, high input power operation at JET has also recently produced examples of ELMs with low frequency (~4Hz), each carrying 1 MJ of plasma stored energy. There is evidence for target ablation and strong impurity influxes during these events, demonstrating a unique JET capability to explore ITER relevant ELM induced erosion. In many cases, large Type I ELMs on JET are also associated with rapid displacements of the divertor strike points, often over several cm. This is of potential importance for ITER with regard to power dispersal. Improved measurements together with modelling on JET, support the conjecture that these jumps may be due to the peeling off of a current carrying plasma layer during the ELM [Solano et al., this conference].

Using configurations optimised for pedestal and divertor diagnosis (DOC-L, DOC-U), a large database describing Type I ELM behaviour and the H-mode pedestal profiles has been compiled at JET. At high ∇ and q_{95}, so-called 'convective' Type I ELMs (ie ELMs for which the decrease in the pedestal pressuredue to the ELM is mainly due to a decrease in temperature) have been found in JET at low pedestal collisionality. If such a regime were also found in ITER, extrapolated ELM target power densities would be within tolerable limits. [Loarte et al., APS 2004 – Phys. Plasmas 11 (2004) 2668].

Regimes with intrinsically small ELMs have been pursued via dimensionless similarity studies with both Alcator C-Mod (EDA H-mode) and ASDEX-U (Type II). Despite considerable effort, it has not yet been possible on JET to reproduce the EDA H-Mode. On the contrary, by operating at low I_p (1.2MA) inorder to obtain high poloidal beta ($\nabla_p > 1.6$), as originally developed on JT-60U, and using a quasi-double-null configuration, pure Type II ELMs have been obtained for the first time on JET, albeit at rather high ∇*values of about 10 ∇ ∇^*_{ITER}. To test this regime at higher currents, power levels are needed exceeding current heating capabilities at JET, and will therefore have to wait until further power upgrades at JET. [#see paper by J Stober] Regimes with intrinsically small ELMs have been pursued via dimensionless similarity studies with both Alcator C-Mod (EDA H-mode) and ASDEX-U (Type II). Despite considerable effort, it has not yet been possible on JET to reproduce the EDA H-Mode. On the contrary, by operating at low I_p (1.2MA) in order to obtain high poloidal beta ($\nabla_p > 1.6$), as originally developed on JT-60U, and using a quasi-double-null configuration, pure Type II ELMs have been obtained for the first time on JET. To test this regime at higher currents, power levels in excess of $$MW are required, and are therefore only possible after further upgrades of the heating power on JET. [#see paper by J Stober]. Active moderation of ELMs by N_2 impurity seeding in plasmas at I_p/Bt = $$/$$ has successfully demonstrated an integrated operational regime using nitrogen seeding by raising the radiated power fraction up to 95% resulting in a Type III H-mode. Confinement is reduced by about 20% but extrapolation of this regime to ITER show that nevertheless Q=10 is possible, by running ITER at 17MA [Rapp ref]. Rapp et al., to be published in J. Nucl. Mater]. Sustained steady conditions up to H_{98}~1, ∇_N=1.8-2.0 for n/n_{GW}=1-1.1 have been separately achieved through argonseeding of Type I ELMing regimes under dual real-time feedback control, using majority and impurity gas inputs as actuators. [#see paper by P Dumortier EPS, P Monier-Garbet at IAEA.]

7. HEATING AND CURRENT DRIVE

7.1. PLASMA ROTATION

Plasma rotation is important for MHD stability and transport. In ITER, the momentum input will be substantially lower than in most present experiments. It is thus of paramount importance to understand the experimental observations of strong plasma co-rotation, despite little momentum input, as in the case of ICRF. A number of theories relate the generation of rotation with the transport of fast particles generated by ICRF [Chan V.S., Chiu S.C., and Omelchenko Y.A., ‘‘RF-driven Radial Current and Plasma Rotation in a Tokamak’’, Phys. Plasmas **9** (2002) 501]. Whereas experiments with co- and counter wave momentum input,allowed to clearly identify the toroidal plasma rotation due to this type of mechanism, they also show thatthis is only a small component overlaid on a larger still unexplained fraction of co-rotation [Eriksson L.-G., Johnson T., Giroud C., Kiptily V., Kirov K., Mantsinen M., Noterdaeme J.-M., Staebler A., Tuccillo A.,Weisen H., and et al., ‘‘Plasma rotation induced by directed waves in the ion cyclotron range of frequencies’’, Phys. Rev. Letters **92** (2004) 235001-1 to 4].

7.2. ICRF HEATING SCENARIOS

Various ICRF scenarios have been studied and/or further optimised to increase our understanding ofthe ICRF physics relevant to ITER.

Fundamental T heating in a trace Tritium plasma has required JET to operate at the lowest frequency (23MHz) and the highest fields (4T). With 1.5MW of ICRH energetic T tails of 80 to 120keV near the optimum fusion cross-section have been produced, increasing the neutron emission by three orders of magnitude. Second harmonic T was shown to be better suited to a plasma with higher T concentrations while the presence of impurities (C or Be) make minority heating of D in H almost impossible since the C6+ impurity, which has the same cyclotron layer as D, influences wave propagation like a much higher equivalent D concentration and directly leads into the mode conversion regime. [Lamalle, this conference].

Minority heating of ^{3}He in H was studied with 5MW of ICRH (3He concentrations varied from 1%to 10%), and is found to be a suitable candidate for the non-activated phase of ITER. The transition from minority heating (^{3}He effective temperature of 100keV, T_e up to 6keV) to mode conversion heating(electron heating, T_e up to 8keV) was sudden and reproducibly near a ^{3}He concentration of 2%. In themode conversion regime, with directed waves, no CD effects were observed. [ref to Lamalle IAEA paper].

An experiment to study pT fusion reactions (p+T-> ^{4}He+n) has been performed during the TTE campaign at JET. Hydrogen minority heating was used to produce energetic proton tails, causing neutron emission from the pT fusion reaction. The observed neutron yield depends on plasma parameters as expected, apart from some discrepancies at very low tritium concentration. The results show the importance of taking into account the neutron contribution from pT fusion reactions while analysing neutron measurements in purely RF heated experiments in tritiated plasmas. [Santala, EPS 2004]

7.3. ICRH CONJUGATE T-MATCHING

The use of 3 dB couplers allow a reliable operation of ICRF generators in the presence of strongload variations such as large sawteeth and ELMs. [Noterdaeme Matching to ELMy plasmas, SOFT 2004, Venice], and will be implemented on half of the JET antennas in 2005. Conjugate T-matching is another method that promises to maintain the power to the plasma even during the ELM coupling transients and is atthe basis of the JET-EP antenna concept [Durodie, SOFT 2004]. Proof of principle tests using a temporary set-up connecting two adjacent straps of the existing A2 antennas in an external conjugate T-configuration showed this to be indeed a viable method to cope with the ELMs [Monakhov, SOFT 2004]. **$$$ MORE DETAILS $$$**

7.4. LHCD COUPLING

LHCD has been a key tool in the further development of ITB plasmas on JET, and substantial progress has been made in view of ITER. Using a recently modified gas pipe to optimise the gas flow near the launcher, thereby improving the deposition of the gas in the region magnetically connected to the LH launcher, up to 2.5MW of LH power was coupled at an ITER relevant distance of 10 cm with a reflection coefficient < 8%, even during large amplitude Type I ELMs. On ELMs with a smaller amplitude, up to 3MW of LH power has been coupled at a plasma distance of 11cm. These results have to be compared to1MW of LH power coupled without gas injection. This has been obtained with D_2 injection, therefore rendering the use of CD 4 (which is questionable in combination with T because of concerns of T co-deposition) no longer mandatory for good coupling results [ref Mailloux, Ekedahl EPS or PRL]

8. DIAGNOSTICS

8.1. EDGE DIAGNOSTICS

Particular efforts have been devoted to improve the spatial resolution of the various measurements inthe plasma core, doubling the ECE channels [De La Luna et al., accepted for Rev Sci. Inst. Vol. 75 (2004)] and increasing the accuracy of the active spectroscopy by increasing the energy of the neutral injector used for the MSE diagnostic from 80keV to 130keV. Particular attention has also been devoted to the edge, which in general is a difficult plasma region for measurements. The Li beam diagnostic has been enhancedincreasing the energy of the beam atoms from 30 to 60 keV and the current from 0.2mA to 1mA [Korotkov, Rev. Sci. Instr Vol 75, 2590-2602 (2004)], enhancing the penetration by at least 50% in ELMy H-Mode plasmas, and allowing the measurement of the electron profile in the region of strong gradients with a spatial resolution of 1cm and an absolute accuracy of 3%. Both the response times of the fast digitizer and the detectors of the edge LIDAR system have been shortened, resulting in a doubling of the global spatial resolution (from 12cm to 6cm) [M. Kempenaars et al, Rev. Sci. Instr, 75, (oct 2004)]. Together with the development of special plasma configurations (DOC-L and DOC-U) which are designed to maximise the number of simultaneous plasma edge measurements, these enhancements have allowed a much more detailed study of the gradients of electron temperature and density in the edge, and a better documentation of the nature of the ELMs. (See Section 5.4). [Loarte, this conference].

8.2. BURNING PLASMA DIAGNOSTICS

Significant progress has been obtained in the measurement of the 3.5MeV alphas and the 14MeV neutrons, and a direct measurement of all products of the D-T reaction in JET is now possible. Gamma ray spectroscopy now provides almost routinely the spatial distributions of the fast particles (fast D and He ions accelerated by ICRF, or 3.5MeV He particles from the fusion reaction). The technique is based on the detection of gamma rays from the nuclear reactions $^{9}\mathrm{Be}(\nabla,\mathrm{n}\nabla)^{12}\mathrm{C}$ and $^{12}\mathrm{C}(\mathrm{d},\mathrm{p}\nabla)^{13}\mathrm{C}$ which both have a high energy threshold for the alpha (E > 1.7MeV) resp. D (E> 0.8MeV) particle involved. The gamma rays are sufficiently separated in energy (4.44MeV, resp. 3.01MeV) to allow a good discrimination between both, thus allowing a simultaneous measurement (Fig. 5). Moreover, by observing the time dependence of the gamma ray emissions, the confinement of the fast alphas could be directly observed for the first time in atokamak in various plasma regimes (See Section 5.1). The technique has been extended to the detection of fast 3He and H particles, using resp. the reactions $^{12}\mathrm{C}(^{3}\mathrm{He},\mathrm{p}\nabla)^{14}\mathrm{N}$ (gamma emission at 4.78MeV) and $^{12}\mathrm{C}(\mathrm{p},\mathrm{p}'\nabla)^{12}\mathrm{C}$ (gamma emission at 4.44 MeV). Further development of the gamma-ray diagnostics towards a higher time resolution and operations with neutron filters in the presence of large fluxes of DT neutrons (as will be required for ITER) is being considered for future JET operations.

For the study of lost fast particles, a new diagnostic based on activation of suitable samples depending on the energy of the fast particles to be detected (fluor-titanium compounds) has been developed, and has allowed to confirm the theoretically predicated dependency of the fast losses on the poloidal angle.[G. Bonheure, Rev.Sci Instr. 75, 3540 (2004)]. The technique has the potential to discriminate between the losses of various fast particles, and will be further pursued in future campaigns.

The Magnetic Proton recoil Spectrometer has been further refined to also allow absolute measurements of the total 14MeV neutron yield. The results of the spectrometer are in agreement with the other two independent neutron measurements at JET (neutron cameras and the delayed emission samples), and JET is now the only machine in the world having 3 independent measurements of the total neutron yield. Detailed tritium transport studies have been performed with the JET horizontal and vertical neutron cameras and new ITER relevant compact detectors (NE213 14 MeV neutron spectrometer [A. Zimbal et al., Rev. Sci.Instr, 75, 3553 (2004)] and Carbon Vapour Deposited (CVD) diamond counters [M. Angelone et al., EFDA JET Report PR(04)10 (2004); Rev.Sci. Instr, accepted for publication) have been successfully tested,for the first time in a tokamak environment, and offer excellent prospects for use on ITER.

Major progress has been obtained in determining the plasma isotopic composition. The tritium content of the plasma was measured for the first time during TTE using a Neutral Particle Analyser (NPA),especially designed for the determination of the isotopic composition by detecting simultaneously the neutral fluxes of all hydrogen isotopes leaving the plasma at various energies and therefore from different radial positions. [36-SOFT]. This diagnostic is particularly effective for neutrals born in the outer regions of the plasma, where it complements the core measurements of the neutron cameras.

The study of Alfvén Cascades excited with fast ions in reversed magnetic shear plasmas has been greatly facilitated using a novel microwave interferometry technique. The technique measures the density perturbations associated with ACs. This technique is much more sensitive to their presence than detection with pickup coils, because the density perturbation contributes twice along the path of the microwaves through the plasma, while external pickup coils detects only those modes for which the (low amplitude) tail can reach the coil in the plasma edge. An illustration of the results obtained with the new technique clearly shows the unprecedented frequency and time resolution compared to the conventional technique . [Sharapovet al., PRL 93, 165001/1-4].

CONCLUSIONS

- H-mode high density high confinement operation confirmed. Increases confidence in ITER Q=10 objective
- H-mode Beta scaling could be more favourable than foreseen in standard ITER scaling (requires confirmation with further experiments at higher power, filed and current) and density peaking occurring at low nu* => could lead to higher fusion performance on ITER
- Hybrid regime looks promising for long pulse operation on ITER, but scaling remains to bedetermined (high beta, good confinement, acceptable edge conditions simultaneously) / requires further experiments at lowest possible ∇* in relevant range of ∇* and edge conditions. Some progress can be expected on JET with the on-going power upgrade (full NB remains to be exploited, a couple more MW of NB expected, several more MW of ICRH with conjugate-T, ITER-like antenna in 2006), but further power upgrade would be valuable
- AT promising but scaling remains to be determined. RWM studies indicate that ITER should be equipped with adequate coils for high beta operation
- Deposition of energy during disruptions seems to be less constraining than expected for the ITERdivertor, but could be difficult for Be wall. Further data needed. Effects on metallic wall

would require relevant wall conditions.

- Erosion seems to affect mostly the main chamber. Good news for ITER w.r.t. T retention, although erosion in divertor is not excluded. Metallic wall might nevertheless be needed. Flows inSOL remain a puzzle for modelling / extrapolation to ITER remains a challenge.
- ELMs remain a significant challenge for ITER. Mild type II ELMs obtained on JET, but only atlow current, i.e. very high collisionality, as on smaller machine / question of wether such regimes exist at relevant collisionality remains to be assessed. Therefore type III ELMs should not be ruled out for use on ITER and their applicability to ITER further explored. ELM mitigation techniques (as, e.g. by using pellets as on ASDEX-Upgrade) should be explored.

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[1]See complete list of the contributors in the Appendix