Transport and Boundary Physics:
Summary Review

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Abstract. In this paper the contributions presented at the 18th IAEA Fusion Energy Conference in the field of transport and boundary physics will be summarised with reference to the following distinct issues: H-mode physics, Internal Transport Barrier formation, transport studies, Radiative Improved modes and impurity seeding, divertor and He exhaust, new configurations.

1. Introduction

In this paper the contributions presented at the 18th IAEA Fusion Energy Conference in the field of transport and boundary physics will be summarised. More than 90 papers have been presented at the conference which are related to transport. In the following they will be analysed with reference to the following distinct issues: H-mode physics, Internal Transport Barrier formation, transport studies, Radiative Improved modes and impurity seeding, divertor and He exhaust, new configurations. The results on the internal transport barrier behaviour are considered also in Ref.[1].

2. H-mode physics

ITER operation require the achievement of a global confinement time in line with the ITER98y(2) scaling at a Greenwald number $G \equiv n/n_{GW}=0.85$ and $q_{95}=3$, with n_{GW} being the Greenwald density. The simultaneous achievement of all these conditions is still a challenging objective but important progresses were reported at the conference.

In order to obtain H-mode confinement the L-H threshold power P_{LH} must be exceeded by a significant amount: a factor 1.8 is considered a safe margin although at high triangularity the power required decreases up to 1.4 P_{LH}, as shown by JET [2].

To maintain H-mode confinement at high density is still an issue. High Field Side pellet injection on JET allowed to reach Greenwald numbers up to G=1.6 [3] but degradation in confinement is observed above G=0.8. Upon precisely timing the injection of pellets it was possible to obtain H97=0.9 with G≈0.9. Low field side pellet injection with shallow pellet penetration triggers the L-H transition at low heating power in DIIID [4]. Encouraging results were obtained on DIIID by simultaneous gas puffing and strong divertor pumping in 1.2MA discharges [5]. The ELM amplitude is reduced to 20%. In these conditions G=1.4 is achieved likely because the gas injection rate nearly balance pumping and the convected parallel component of the flow in the Scrape Off Layer is large. With a high parallel convected component, the X-point temperature remains high and density low, preventing the formation of a cold radiative mantle at the X point The pedestal density saturates at G=0.9 and density peaking is observed (n/n_{line} increases from 1.3 to 1.5). The highest density discharges are terminated by the onset of a 3/2 neoclassical tearing mode. H-mode confinement at the Greenwald limit can be maintained on JET with intense auxiliary heating ($P/P_{LH} \approx 2$) and is obtained with small ELM activity [6]. The power required to access this regime decreases with triangularity.

The achievement of H-mode confinement is observed with all the heating methods. H-mode confinement with Lower Hybrid Current Drive was reported on HT7 [7].

2.1 Global confinement scaling

Present scaling laws for the energy confinement time predict a large dependence of the energy confinement time on plasma shaping. This has a significant impact on the ITER confinement predictions. The effect of shaping on confinement was addressed by JET in a series of dedicated scans at fixed plasma current and edge safety factor. A strong dependence on elongation has been observed, mainly related to the effect on the pedestal density, with a weak dependence on triangularity[8]. The increase of the energy confinement with triangularity is observed on DIIID at low to moderate density, whereas weak dependence is obtained at high-density [9].

2.2 L-H transition and pedestal scaling

Although the H-mode is universally observed in tokamaks with a magnetic separatrix and the qualitative understanding of the L-H transition mechanism is satisfactory, the quantitative prediction of the L-H transition has still major uncertainties. New data, which compare different divertor geometries, were presented at the conference. The role of the edge plasma parameters at the L-H transition has been further investigated. Nevertheless, the scaling of the pedestal characteristics with dimensionless parameters is still lacking and this makes difficult the extrapolation to burning plasma experiment.

Present L-H power threshold data have been collected from experiments with very different divertor geometry. The possible role of divertor geometry has been suggested as the cause of the large uncertainty in the ITER threshold power. A possible support to this hypothesis was produced by JT60-U, which reported a reduction of P_{LH} with the W-shaped closed divertor geometry [10]. The density dependence of the power threshold was stronger than the open divertor case.

A systematic study of the dependence of P_{LH} on the change in the divertor geometry was provided by JET[11]. A reduction of the power threshold with the decrease in the X-point height is clearly shown, whereas the threshold power sharply increases below a plasma-outer wall distance of 3cm.

The role of the radial electric field in the L-H transition has been further elucidated. High-resolution studies were performed on JFT2-M using the Heavy Ion Beam probe [12]. The density and temperature fluctuations are generally suppressed at the time of the edge potential drop. Such a drop occurs on a fast time scale (≈10-100μs) if the input power is above threshold. ExB turbulence suppression on TEXTOR was investigated by using external electrodes[13]. Turbulence measurements in the thin shearing layer show a reduction of the turbulent particle flux even at low radial electric field strength. Fast current ramp up/down and magnetic compression were used to control the edge radial electric field and the L-H and H-L transition on TUMAN 3M [14]. Progress in turbulence simulations of the X-point dynamics during the L-H transition has been reported [15]. A correlation between the dependence on magnetic geometry of the L-H threshold and the characteristics of the resistive X-point mode has been pointed out, which might be in agreement with edge turbulence measurements on DIIID.

The existence of a critical edge temperature for the onset of the L-H transition has been proposed in the past on the basis of the experimental evidence of ASDEX-U. However, no evidence of a critical edge temperature has been obtained by pellet injection on DIIID [16]. In these conditions the edge temperature drops to very low values before the transition, following the injection of pellets. The data support the key role of quantities such as T_e/T_e and p_e/n_e at the pedestal location.

H-mode performance in a next step experiment depends on the extrapolation of the H-mode barrier height and width. On DIIID [16], the barrier height increases with triangularity, due to better MHD stability properties, whereas the barrier width is weakly dependent on triangularity and is well described by a relation of the form $\Delta \approx \beta_p{}^{1/2}$, whereas no dependence on the ion gyroradius is found. A strong dependence of the pedestal contribution to the global confinement was reported by JET.

3. Internal transport barrier regimes

The ITER requirements for Advanced Tokamak (AT) operation correspond to H98=1.5 at G=0.92, 70% bootstrap fraction, β_N=3.5 and q_{95}=4.2 and can be accessed with formation of an internal transport barriers (ITBs). Although the formation of ITBs is a well-known fact, the control of these regimes in order to reach steady state conditions for ITER relevant parameters is still an issue. Nevertheless, significant progresses were reported at the conference by various groups. More details can be found in Ref. [1].

On DIIID, a product β_NH89=10 for 5 a duration of 5τ_E (with τ_E the energy confinement time) has been achieved at q_{95}=5 and β_NH89=7 for a duration 34τ_E[17]. Also JET reported in the past the achievement of similar values of this figure of merit (β_NH89=7) at relatively low q_{95} operation, although for a shorter duration ($\approx$3τ_E). At this conference record value have been also reported by JT60-U (β_NH89=7.2 for 6τ_E [18]) and ASDEX-U (β_NH89=7.2 for 10τ_E [19]).

3.1 ITB access an control

The ITB formation is the result of two simultaneous effects: a reduction of the growth rate (due e.g. to the formation of low magnetic shear profiles or to large Shafranov shifts of the equilibrium magnetic surfaces) and the production of a sheared ExB flow.

Weak magnetic shear condition are usually obtained by fast current ramp and strong electron heating, but such a scenario is only transient and needs a way to maintain a hollow current density profile by non-inductive methods. The formation of radial electric fields can be obtained by controlling the plasma rotation. A large ion pressure gradient, such as that characterising an ITB, typically helps in maintaining a sheared ExB flow.

The results presented at the conference generally support this picture, although the capability of a quantitative prediction of the ITB formation conditions is still lacking.
Results from JET show that a significant decrease of the threshold power for ITB access with LH preheat can be obtained [6], probably due to a more efficient formation of a low magnetic shear region. ITB formation with LH heating was reported by FT2 [20].

The formation of a reduced core transport region by intense ECRH during the current ramp was reported on FTU [21], with a central temperature of 14keV, and by localised ECRH/ECCD on DIIID [22] and JT60-U [23].

Control of the ExB shearing rate with various methods was reported. DIIID was successful in controlling the radial extension of the ITB and its gradient by counter NBI [24]. The control of toroidal rotation by co/counter NBI and the change in the power deposition profile to control p_i were shown to be effective on JT60-U for controlling the ITB behaviour [25]. The formation of an ITB on the density profile by ICRH was reported by C-MOD [26].

Finally, the correlation between the formation of ITBs and the onset of sheared ExB flows has been reported also by stellarators. Evidence of significant ExB flow associated with the presence of rational surfaces has been reported by TJ-II [27]. The measured properties of the turbulent fluctuations are observed to change accordingly. Three main bifurcated branches have been observed on CHS [28] in low density ECR heated plasmas by Heavy Ion Beam Probe measurements. The three branches (hill, dome and bell) are characterised by the formation of internal transport barriers.

3.2 Quiescent modes

The ELMy H-mode is the confinement regime with the largest energy confinement. However, the MHD relaxations associated with Type-I ELMs make this regime unfeasible for a burning plasma experiment due to the associated divertor erosion. Several new regimes characterised by a quiescent boundary were reported which do not suffer from impurity accumulation as the ELM-free H-mode thanks to a quasi-coherent MHD activity. These regimes include:

- The Enhanced D_{α} mode (EDA) observed on C-MOD [29] at medium to high density and high triangularity (δ>0.35); confinement is about 85% the ELM-free confinement. Impurity accumulation is avoided by continuous MHD activity at the plasma edge.
- The quiescent H-mode (QH) and the quiescent double barrier mode (QDB) observed in DIIID [24] (β_NH89=7 for 5τ_E). The optimal ITB profiles have small gradient and large ITB radius ($r_{ITB}/a \approx 0.7$-0.8). The ITB location depends on both magnetic and rotational shear and not only on the magnetic shear: fast current-ramp experiments, which bring the q_{min} position from 0.4a to 0.9a, do not move the ITB location. The amount of rotational shear depends on the interplay between the effect of the pressure gradient and the toroidal velocity shear. Counter neutral beam injection makes dominant the ExB rotation associated with the pressure gradient and facilitates the expansion of the ITB radius. The QH mode is characterised by an ELM-free edge with MHD activity which allows density and impurity control for more than twenty energy confinement times with divertor pumping.
- The Type II ELMs regime observed in ASDEX-U [30]; contrary to earlier observation of this regime on DIIID and JT60-U which occurred at medium density, the Type-II ELM regime is observed at G=0.8 on ASDEX-U. Also in this case a high triangularity is required. The energy confinement in this regime is close to the ITER98y2 scaling and significantly larger than that observed with Type-III ELMs. The peak power load is significantly reduced from 5MW/m^2, in the case of Type-I ELMs to 2MW/m^2.

It should be noted that generally, regimes with ITB have a low ELM activity as shown by JET and JT60-U.

Also non-tokamak devices have been able to achieve improved confinement modes by controlling the turbulence behaviour: results were presented by RFX (quasi-single helicity state [31]) and by MST [32] and TPE-RX [33] (by pulsed poloidal current drive). Control of MHD modes by rotating helical fields has been reported by STE-2 RFP [34]. The use of current profile modifications, in order to obtain current density profiles which are stable against tearing modes, lead to an energy confinement improvement of four times and to a particle confinement improvement of eight times the values without pulsed poloidal current drive.

4. Transport studies

Most of the transport studies have dealt with the issue of profile stiffness and understanding of the Bohm/Gyrobohm scaling of tokamak transport. A related issue is the investigation of the role of zonal flows in regulating the turbulence dynamics. Global gyro-fluid and gyro-kinetic simulations have proven to be essential in helping the theoretical modelling and the experimental analysis and were extensively reported at the conference.

4.1 Critical gradients and profile stiffness

Tokamaks are known to exhibit resiliency in the electron temperature profile. The origin of profile stiffness has been discussed for long time and attributed to different causes such as global MHD stability. More recently, attention has focussed on the electron temperature gradient (ETG) mode, a very short wavelength instability ($k_\theta\rho_i$>>1) characterised by a threshold in the parameter L_{Te}/R, with L_{Te} being the characteristic scale length of the electron temperature profile.

The results presented at the conference show that resiliency is observed both on T_e and T_i but not on density on ASDEX-U [30]; such a resiliency is found in most of the operational window. Profile stiffness leads to a correlation between core and edge temperature: the increase in triangularity, which allows larger edge temperature gradients, lead to a reduction of transport in the plasma core. The analysis of ECH discharges shows that the electron thermal conductivity has a “universal” behaviour as a function of T_e/T_e, in agreement with the ETG model.

On FTU [35], T_e resiliency is observed only in the presence of a finite magnetic shear. Discharges with strong ECRH on the current ramp [21] show a diffusive behaviour of the electron transport in the region where the magnetic shear is low or negative, even at large temperature values (T_{eo}≈14keV) which are expected to imply a stronger stiffness. On the contrary, low temperature discharges with monotonic q profiles clearly show indication of profile resiliency.

Transient transport studies generally provide a useful tool for pointing out non-diffusive behaviour related to profile stiffness. On RTP [36], a convective inward term is needed in order to simulate the electron temperature profile in ECH dominated plasmas. Transient transport studies in ASDEX-U [37] performed with laser blow-off techniques and ECRH modulation, in order to provide a negative/positive temperature perturbation of the edge, also show non diffusive behaviour, although the agreement with models such as the IFS-PPPL is not satisfactory.

JT60-U discharges have been used to test transport models characterised by different “stiffness”. The IFS-PPPL model reproduces the ion temperature within 10% whereas the electron temperature shows larger discrepancy. The multi-mode and RLWB model yield typically too high core temperatures [38].

The proximity to ETG-mode marginal stability was tested on Tore Supra [39] by Fast Wave Electron Heating in a large range of heating power. The selected discharges had a hot electron plasma (T_e>$2T_i$), in order to minimise the electron-ion coupling. No fast particle population was present, all the wave damping being on the electrons. The density and safety factor profiles were kept fixed and the FW power was varied between 1.5MW and 7.4MW. The experimental data show clear evidence of profile stiffness, with the normalised critical electron temperature gradient being a linear function of s/q, with s the magnetic shear.

Several theoretical analyses pointed out that ETG turbulence is less affected by non-linear saturation mechanisms than ITG turbulence and this allows the mode to reach a saturated amplitude larger than that predicted by usual mixing-length type arguments [40-42].

Finally, also non-tokamak systems observe a resiliency, as shown by high-resolution Thomson scattering measurements in LHD [43], where triangular-shape profiles are observed.

4.2 Bohm-gyrobohm scaling

Tokamak confinement exhibits a scaling which is either Bohm-like or gyro-bohm depending on the operational regime. The ITER H-mode scaling is gyro-bohm and energy confinement in stellarators also exhibit gyro-bohm features (with LHD [44] reporting an increase up to a factor 2 over the ISS95 scaling due to the pedestal contribution).

In order to account for long radial correlation length turbulence, full-torus simulations are needed. Earlier gyrokinetic numerical simulations in full-torus geometry, which neglected the effect of zonal flows, predicted Bohm-like transport and long radial correlation length turbulence. Recent gyrokinetic simulations, which include the effect of zonal flow, still predict Bohm transport although turbulence exhibits gyro-bohm behaviour [45]. Such a discrepancy is due to the numerically found saturation level which correspond to $v_{ExB} \approx v_{dia}/\rho_*^{1/2}$, instead of the classical mixing-length estimate $v_{ExB} \approx v_{dia}$. A possible explanation for such a surprising behaviour might be a sharp variation of the diamagnetic velocity profile.

Very careful measurements by beam emission spectroscopy of long wavelength fluctuations were performed on DIIID [46] to investigate this issue, by making ρ_* scans with the other dimensionless parameters being kept fixed. The results show a gyro-bohm behaviour of fluctuations: the correlation length scales as ρ_*, the correlation time scales as a/C_s and the density-fluctuation level scales as $\rho_*^{1.4}$. Surprisingly, the scaling of the thermal diffusivity is intermediate between Bohm and gyrobohm. Nevertheless, such an intermediate scaling is in agreement with numerical simulations performed with the GLF23 code, pointing out the existence of mechanisms breaking the intrinsic gyro-bohm scaling such as the proximity to marginal stability for ExB turbulence suppression.

The possible role of long radial correlation length events was also discussed [47-48]. Theoretical paradigms for avalanche phenomena were limited in the past to rather specialized models (sand piles, etc.). Models for the avalanche dynamics for simple models of ITG turbulence were presented at the conference. The model includes the effect of magnetic curvature and convective non-linearities and the self-regulating mechanism associated with a secondary Kelvin-Helmoltz instability, which causes the brake up of strongly anisotropic streamers.

4.3 Zonal flows

Several theoretical papers have been presented on zonal flow dynamics. Zonal flows are nonlinearly driven by turbulent fluctuations and are damped by collisions. Their presence regulates the turbulence level via shear decorrelation. The agreement between the numerical [40-42, 45] and the analytical [49] approach is quite satisfactory near marginal stability: for weak drive the effect of collision is large. Above marginal stability the agreement is more qualitative and the role of zonal flows in affecting turbulence is less clear.

Finally, the role of zonal flows at the core/edge boundary has been investigated [50], finding a large variety of distinct radially and poloidally coherent flow structures.

4.5 Impurity transport

An analysis of the behaviour of the impurity transport in the edge pedestal region has been presented by C-MOD [51]. The poloidal asymmetry of the x-ray emission may be qualitatively explained on the basis of neoclassical theory.

The analysis of impurity transport in FTU [52] during strong ECRH on the current ramp shows that the central particle diffusivity depends on magnetic shear. Very low Mo diffusion coefficients are observed in conditions that are likely to have non-monotonic current density profiles, whereas significant departure from corona equilibrium characterises discharges with monotonic current density profiles.

The particle and impurity transport in DIIID shows a neoclassical behaviour when regions of neoclassical ion thermal transport are produced [53]. Discharges with anomalous ion transport show anomalous particle transport and no tendency to impurity accumulation.

5. RI modes and impurity seeding

Experimental investigation of the Radiative Improved (RI) mode has been performed on several tokamak devices and reported at this conference. The interest of this mode of operation is related with the possibility of operating at high density and high confinement in steady state conditions and with a large radiated fraction. The onset of the RI phase is attributed to a reduction of the ITG growth rate due to the increased dilution. The corresponding reduction of the particle transport increases the density profile gradient, further reducing the turbulent transport.

TEXTOR [54] reported the achievement of improved confinement up to 40% above the Greenwald density limit by gas fuelling rate optimisation in order to avoid the confinement rollover and the transition to L-mode that typically occurs with strong gas puffing. In these condition the beta limit (β_N =2.2), probably related with the onset of neoclassical tearing modes, is reached at G=2.

Impurity (Ne, Ar, Kr) injection in L-mode DIIID discharges [55] leads to an increase of confinement up to a factor two over the ITER89P scaling and a decrease in long wavelength fluctuations. Transport is reduced in all channels with Ne injection: the ion thermal diffusivity decreases to the neoclassical level, the electron diffusivity decreases by a factor $\leq$1.5 and particle and momentum diffusivities are also reduced. Impurity accumulation in the centre brings Z_{eff} from 1.5 to 3.4. A decrease of the turbulent fluctuations in the plasma core is observed with Ne injection. The analysis shows that such a behavior may be correlated with an increase of the ExB shearing rate associated with changes in the toroidal velocity profile.

Argon seeding of ELMy H-mode discharges in JT60-U [56] leads to high radiated power fraction (f_{rad}=80%) and high confinement (consistent with the ITER98y2 scaling) up to 65% of the Greenwald density. The confinement improvement is associated both to the core and the pedestal contribution.

JET [57] reported results obtained in three distinct scenarii: limiter L-mode, divertor L-mode and impurity seeding of high density ELMy H-mode. Limiter L mode discharges reached a radiated fraction f_{rad}=0.7 at 80% of the Greenwald limit but with a limited confinement improvement (H89=1.3) perhaps due to the lack of density profile peaking observed in other devices. Divertor L-mode discharges obtained a transient confinement improvement (up to H89=2) at 40% of the Greenwald limit and frad=0.4. A confinement improvement lasting for three energy confinement times has been obtained in ELMy H mode discharges, following the main gas puffing phase, up to 90% of the Greenwald limit. Discharges with Argon seeding show little increase in Z_{eff} with f_{rad} =0.6. Little evidence of density peaking was found.

6. Divertor, SOL physics and He exhaust

The compatibility of the ELMy H-mode scenario with acceptable divertor operations remains an issue for a next step experiment. Nevertheless, it has been confirmed that He exhaust does not represent a problem for this scenario: values of τ^*_{He}/τ_E as low as 7.6 with Argon frosting of the divertor cryo-pump have been reported by JET [3] and τ^*_{He}/τ_E=3 with both leg divertor pumping has been demonstrated in JT60U [58].

In a series of dedicated pulses in which the strike point position has been changed JET [3] has shown that the deposition at the divertor plates show a strong asymmetry between the outer and the inner divertor with two third of the power flowing to the outer targets.

Net erosion free operation of the graphite covered DIIID divertor [59] have been reported. This result has been achieved by the combination of low temperature plasma in the divertor, which reduces the physical sputtering, and long-term wall conditioning, which reduces chemical sputtering.

Tungsten wall coating of the main chamber was tested in ASDEX-U [60] following the previous experience of W-coating of the divertor. The coated area was about 1.2m^2 corresponding to the lower part of the central column, where charge exchange neutrals are predicted to be dominant. Despite the large coated area the central concentration remained below the detection limit. Further wall coating is expected in the near future.

The W shaped divertor of JT60-U [58] has been modified from inner to both-leg pumping. Good He exhaust both in H-mode and ITB plasmas has been observed: τ^*_{He}/τ_E=3 with attached plasmas and both leg pumping has been achieved in ELMy H mode plasmas with an He exhaust efficiency 40% larger with respect to the inner leg pumping. In reversed shear ITB plasmas a ratio τ^*_{He}/τ_E =8 was measured in the case of high recycling divertor which, however, corresponds to limited confinement improvement.

The open divertor geometry of TCV allows the study of detachment in different magnetic configuration by varying the X-point distance from the plate and therefore the connection length and the flux expansion. The divertor geometry is typically characterised by short X-point distance from the vertical target and a long distance from the horizontal target. A low critical density for detachment of the horizontal target is consequently observed [61].

The island divertor concept has the advantage, on the poloidal diveror concept, to require smaller field perturbations which require small current control coils [62]. Stable high recycling conditions will be demonstrated in the island divertor experiment on W7-AS.

Cross-field transport in the far SOL of C-MOD [63] is large (D_{eff}≈10 D_{Bohm}) and causes a large flux of ions directly to the wall rather than inside the divertor. This effect determines the neutral pressure in the main chamber independently of the divertor geometry.

The TFTR experience shows that Lithium conditioning may be an alternative to boronisation and siliconisation. Test of Lithium Capillary Pore System has been successfully performed in T11M [64].

After the final year of operation of the Tore Supra ergodic divertor a review of the operational flexibility of such a configuration has been presented at the conference [65]. Good compatibility of ICRH and LH coupling has been demonstrated. Despite the large volume of the open field line (≈36%) the plasma current has not been limited and stable, low-q operation have been performed.

Recycling and wall pumping has been studied in long duration (≈70min) discharges in TRIAM-1M [66] sustained by Lower Current drive at 2.45GHz (n_{line}≈0.2 $10^{19}m^{-3}$) and at 8GHz (n_{line}≈$10^{19}m^{-3}$). In the low-density discharges a cyclic behaviour of the recycling coefficient has been reported.

8. New configurations

Preliminary results of several new devices were reported at the conference.

8.1 Spherical tokamaks.

NSTX [67] reported the achievement of 1MA plasma current and the initial results on Coaxial Helicity Injection and High Harmonic Fast Wave heating. MAST [68] reported the achievement of 0.7MA of plasma current and the initial results of NB injection and Electron Cyclotron Heating at the 2nd harmonic. Preliminary results from GLOBUS-M [69] TS2-ST [70] where also presented

8.2 Stellarators

Initial operation of the H-1NF Heliac [71] with increased ECH and ICRF power and higher magnetic field were reported. The L=1/M=4 helical axis heliotron device HELIOTRON J [72] started the operation at the end of 1999. Preliminary results with ECH power up to 0.4MW were reported.

8.3 Electric tokamaks

Initial operation of the large aspect ratio (R/a=5m/1m) Electric Tokamak [73] with ohmic plasmas showed the achievement of densities about twice the Murakami limit and high energy confinement. Preliminary results on ICRF coupling were reported.

8.4 Miscellanea

Compact Tori (CT) injection was successfully performed in JFT2M [74]. At a CT density n_{CT}≤9 $10^{21}m^{-3}$ and velocity u_{CT}≤300km/s the CT is able to penetrate the magnetic field (B=1T) up to the plasma centre.

9. Conclusions

As a conclusion it is worthwhile stressing the main progress achieved since the last IAEA meeting.

For the ITER baseline scenario, i.e. the ELMy H-mode, the achievement of high confinement regimes at density values close or even above the Greenwald limit was reported. A tolerable ELM activity (corresponding to Type II ELMs) has been achieved at relatively high density (n/n_{GW}=0.8) and high confinement (H_{98y2}=0.95), i.e. in conditions close to those required for ITER. Steady H-modes with quiescent boundaries have been also produced. However, the compatibility with divertor operation remains an issue since high-radiated fraction regimes have not been obtained simultaneously with the high-density and high confinement phase. The fact that He accumulation is not a problem, is now a well-established result.

Advanced Tokamak operation show a continuous improvement especially in the duration of the high beta and high confinement phase. Regimes with quiescent transport barriers have been reported which might avoid the problems related with plasma control and divertor compatibility. The control of the ITB location and width is becoming a real possibility, although it is unclear whether the presently used tools will be effective also for a burning plasma experiment. The achievement of high-density regimes and low edge safety factors ($q_{95}\approx4$) is still an issue. A problem is also the control of impurity accumulation inside the internal transport barrier.

In the understanding, a general paradigm for transport barrier formation is emerging. The combined effect of weakening of the turbulence drive (via reduced magnetic shear, finite Shafranov shift, impurity injection) and ExB shearing is receiving growing supporting evidence. At this conference the control of the location and of the width of the transport barrier, based on this picture, has been reported.

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