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ABSTRACT

As reported at the FEC 2012, operation with a Be/W wall (JET-ILW) at JET has an impact on plasma confinement and scenario development relative to the carbon wall (JET-C). The main differences observed are a degradation of confinement for low β_N scenarios (typically $H_{98(y,2)}$~0.8) and W accumulation in the plasma core at low fuelling gas rates, in some cases leading to an increased core radiation causing the triggering of MHD followed by a disruption. During the JET 2014 campaign, both high and low β_N scenarios started to be successfully developed towards high plasma current where control of W accumulation in the plasma core and methods to reduce the plasma temperature near the target plates were successfully integrated in the scenarios. Plasma energy confinement has been improved at $I_p < 3$MA.

1. INTRODUCTION

Operation with the Be/W wall at JET has had an impact on plasma confinement and the scenario development relative to the carbon wall (JET-C) as reported at the Fusion Energy Conference (FEC) 2012 for the initial ILW campaigns from 2011-2012 [1]. The main differences observed were a degradation of confinement for baseline like scenarios (β_N ~1.5-2 with fully diffused current profile) typically $H_{98(y,2)}$ ~0.8 and W accumulation in the plasma core at low gas fuelling, in some cases leading to an increased core radiation causing the triggering of MHD followed by a disruption. In order to develop stationary high performance plasmas in DD and ultimately in DT, subsequent effort has focused on understanding and improving the energy confinement on the control of W accumulation in the core and robustness of the scenarios against disruptions. This paper presents the progress of high performance operation in DD towards a future DT campaign, including the integration of technical constraints such as energy and temperature limits of plasma facing components (PFCs) due to normal plasma power loads and disruptions. Baselines scenarios have been developed up to 4MA in quasi-stationary conditions, avoiding W accumulation and controlling the divertor heat loads. These employ significant gas puffing and further improvements to performance [2] may be achievable. The typical values of $H_{98(y,2)}$ achieved are ~0.7-1.1 quasi-stationary with the best confinement at low plasma currents ($I_p < 2.5$MA). Hybrids scenarios (tailored q profile), aiming at high β_N, (~2.5-3) have been developed up to 2.8MA with the minimum gas fuelling needed to control W accumulation. These high β_N plasmas can reach $H_{98(y,2)}$ ~1.2-1.4 [3]. The paper is organised as follows: section 2.1 optimisation of ICRH for W accumulation control, 2.2 control of the heat load and mitigation of disruptions, 3 confinement behaviour and section 4 summary and future prospects including DT.

2. OVERVIEW OF SCENARIO DEVELOPMENT

Operation with a beryllium (Be) wall and tungsten (W) divertor at JET is restricted to a maximum surface temperature for Be of 950°C, to a maximum surface temperature of the bulk W (divertor) of 1000°C to limit thermal fatigue and to 1200°C on the W coated CFC tiles to prevent carbidisation which embrittles the coating. In addition, low W concentrations (c_W<10^{-4}) are required to avoid

excessive core radiation [4]. The W observed in the diverted plasmas is due to W sputtering by Be ions from the main chamber in-between ELMs and by D_2 and Be during an ELM, and depends strongly on the plasma temperature near the target plates [5]. The development of a high performance scenario involves operation at high plasma current (I_p >3MA) and field (B_T >3T) hence high plasma energy and temperature. In order to overcome these limitations, several issues are addressed when developing these scenarios.

2.1. CONTROL OF CORE W ACCUMULATION

Core W accumulation is detrimental in a fusion reactor not only because it increases the minimum triple product for ignition but also because it leads to radiative collapse or in extreme cases to disruptions. In present experiments, in Deuterium, it has been observed that slow W accumulation is consistent with density peaking and the decrease of the core and edge plasma temperature. W accumulation can be controlled either by gas puffing or by increasing the plasma core temperature using ICRH heating [6, 7, 8, 9]. Experiments show that gas puffing reduces the plasma energy (confinement), ICRH heating with hydrogen minority (H) is then the preferred way to control W accumulation, although even in these plasmas a minimum gas fuelling is required to maintain a minimum ELM frequency [10], necessary to avoid W to reach the plasma core. Pellet pacing is an alternative to gas fuelling, but the system was unreliable for these experiments. Additionally, high-Z dust particles or debris can reach the plasma core in very fast timescales causing a sudden collapse of the plasma core temperature. In these cases, high ELM frequency, and/or high additional heating power have higher probability of survival and ultimately recover.

The resonance position for ICRH was optimised for high B_T >3T operation at JET. At high values of B_T, for the available ICRH frequencies using H minority, the resonance layer is off-axis. Moreover, limitations to the ELM resilience of the ICRH systems imply that the maximum power is not always available for on-axis heating. Experiments on sawtooth control in L-mode plasmas, show that having the resonance layer on the low-field side, within the q = 1 surface, can lead to an increase of the sawtooth period which is beneficial to expel impurities in the core as it was demonstrated in [11] and suggested by [4]. A similar experiment, now in H-mode, was done for the low β_N scenario to extrapolate to high values of BT on how off-axis electron heating and a change on the sawtooth period would contribute to W control. ICRH heating on-axis, off-axis and mixed on-/off-axis with hydrogen minority as well as a off-axis heating with He^3 minority was studied. Whilst ICRH with H minority predominantly heats the electrons, ICRH with He^3 minority heats mainly the ions hence indirectly electrons. A clear result is the higher heating efficiency for the case where the position of the resonance layer is inside the q = 1 surface whilst controlling W accumulation. Outside the q = 1 surface, ICRH for W control is not effective. As for the control of the sawtooth period heating off-axis had no apparent benefit. ICRH heating on-axis with He^3 minority (2–7%) was considered and a comparison with off-axis heating with H minority was performed. Figure 1 shows a comparison between H (red) and He^3 minority (blue – 2-3% He^3, magenta – 7% He^3) impact on W accumulation. In the case of He^3 minority, a small increase of W accumulation is observed

in both the SXR and the bolometry signals for the low concentration case, while a large increase is observed for the case with higher He^3 concentration. Moreover, a decrease of stored energy is observed when He concentration above ~7%. This tendency is also confirmed by the neutron rate of these plasmas. The effect of He^3 on confinement and its impact on fusion power is shown in figure 10b) shows the neutron rate for the high performance plasmas as well as for the discharges where He was injected for ICRH heating (<7%). Due to a gas line contamination, He^4 was unintentionally injected into the plasma reaching 8–20% concentration. The presence of He^3 in the plasma will dilute the fuel and significantly increase the energy loss by Bremsstrahlung radiation. This effect is observed in figure 10 showing the total neutron rate for plasmas where the He content is below 7% and for plasmas where the He plasma content goes from 7–20%. For plasmas with He content <2–7%, no clear impact of performance degradation is observed. However, for He content above ~10%, a degradation of the plasma performance is noticeable. To note that although the He affects the neutron production, the plasma stored energy is similar for similar plasmas with and without He for concentrations below 8%. Based on these results, it was decided to use H minority for the development of the scenarios.

2.2. PLASMA TERMINATION

Disruptions are a major concern for the development of high performance plasmas at high plasma current. The rate of disruptions in the high performance scenarios is ~18% in the low β_N scenario (herein "baseline"), and ~20% in the high β_N scenario (herein "hybrid"). The disruption forces scale with I_p^2 and the total plasma energy is large enough to cause melting of the Be PFCs if the plasma becomes in contact with the wall during the disruption. New configurations with low intrinsic force have been developed for the main heating phase, and for the termination to reduce vessel displacement and forces for the worst case, i.e. the disruption occurs at maximum plasma current. As shown in figure 2, a mitigated disruption at 3.5MA is ~330 tonnes while at 2.5MA is ~200 tonnes where the maximum allowed at JET is 850 tonnes, set by low cycle failure of the JET vessel. This sets a maximum I_p<4.7MA even for mitigated disruptions for these configurations. Whilst a disruption might be inevitable, it is still important to reduce the plasma current and energy as fast as possible whilst avoiding wall contact. Alternative terminations ("stops") have been designed which can be triggered if an alarm is raised. Such alarms can be raised on a wide range of real time signals such as the divertor target temperature, wall proximity, or MHD mode amplitude or the increase of the core radiation.

The detection of dangerous conditions and the optimisation of the "stop" responses maximises the probability of soft landing [12, 13]. However, if this is not successful, the massive gas injection (MGI) is triggered injecting a mixture of 90%D_2 +10%Ar to increase radiation and decrease the current quench timescale, The use of MGI is mandatory whenever the plasma total energy is sufficient to melt beryllium in case the disruption involves loss of vertical position (VDE). Figure 3 shows how the accumulation of W can play a strong role even in the planned termination of a discharge. There are impurity events visible on core radiation (box 3) at ~8.8s and ~10.2s but the W concentration (box 6) recovers to well below 10^{-4} during the main heating and sawteeth continue

(box 5). However, after the NBI is switched off (~11.3s), ELMs cease, W increases both in the edge and core. Box 4 shows the ratio of temperature on axis to that at mid-radius, this first flattens and then shows that the temperature profiles become hollow after the sawteeth cease. The current profile often becomes hollow until the reconnection event, whereupon the large mode amplitude triggers the MGI.

2.3. DIVERTOR HEAT LOAD CONTROL

The effect of divertor geometry on confinement with the JET-ILW wall has a stronger impact than with the JET-C wall. For the ITER-like wall, the strong effect on energy confinement is linked to the closeness of the inner and outer strike to the pump throat as shown in figure 4 ($I_p = 2.5$MA). The best performance is obtained when the strike points are close to the pump throat, where pumping is increased, allowing the access to low pedestal density, hence higher pedestal temperature which improve the core confinement through profile stiffness. This effect is discussed in more detail in [16, 17]. In order to achieve maximum performance, the radial position chosen for the outer strike is close to the pump throat, on W coated CFC tiles. To avoid damage of the coatings, the control of (or protection against) heat loads is essential and operational limits on the surface temperature are imposed restricting the pulse length of the discharge. As the additional input power increases (with Ip), the flat-top for the baseline scenario reduces. From a total of 143 discharges, 61 terminated prematurely, of which 16.8% were caused by reaching the temperature limit (1000°C) of the divertor tiles. For the hybrid scenario these times are even more restrictive as this scenario operates at low density, low plasma current and high additional power. In order to progress to higher plasma currents hence higher additional heating power and extend the pulse length (>5s), it is mandatory to reduce the plasma temperature near the target plates. Radial sweeping of the strike point at 4Hz and 4cm amplitude, within the W coated tiles, was integrated in the scenario in all discharges. The results show that sweeping is effective in reducing the divertor tile surface temperature but not sufficient to extend the discharge to 5s.

To decrease further the plasma temperature near the target places, experiments where Ne is injected aiming at increasing the divertor radiation have started. The initial results are shown in figure 5 for 3 pulses where Pulse No: 87410 (red) has Ne and sweeping of the strike point, Pulse No: 87411 (blue) has Ne and static strike point position and Pulse No: 87412 (magenta) has sweeping but no Ne injection. Only a small benefit on cooling the surface temperature with Ne is observed, although in this case, the additional input power is also lower. In this experiment, no degradation of plasma confinement is observed with sweeping or with Ne, and the relative change of Z_{eff} between the seeded and non-seeded cases is ~0.06. The levels of Ne and D_2 were also varied but for low input power, all discharges behaved in a similar way to Pulse No: 87216 (shown in figure 6) where a transition from type I ELMs to ELM-free phases is seen. In stationary type I ELMy H-modes, more experiments are necessary to assess the feasibility of Ne as radiator to reduce the plasma temperature near the target plates. Additionally, the radial length of the sweeping can also be extended towards the bulk W tile in order to further reduce the target surface temperature.

3. CONFINEMENT

Experiments in CFC towards low ρ* and low ν* showed $H_{98(y,2)}$ ~0.9 for plasma currents up to 4.5MA whilst for the ILW $H_{98(y,2)}$ ~0.6-0.8 is typically obtained at plasma currents of 3MA and above. The reduction in confinement at I_p >3.8MA for JET-C, was due in part to the high levels of gas fuelling required to mitigate the ELM size to <1MJ, and comparable to the requirement for gas fuelling with JET-ILW at any I_p [19]. Comparing the thermal energy with that expected from the IPB98(y,2) scaling it is clear that with the ILW, and for plasma currents above 2.5MA, a significant decrease of the thermal energy is observed as shown in figure 7a). For both scenarios, the pedestal mainly determines the thermal confinement, with a contribution to the thermal confinement of ~40% for both JET-C and JET-ILW [20].

Figure 7 (b) shows that higher pedestal pressures are obtained t higher I_p on JET-ILW, (the constant pressure curves (magenta, blue, green and red) guide the eye through the 2, 2.5, 3 and 3.5MA data. However, much higher pressures were obtained on CFC, much higher pedestal temperature being obtained for similar pedestal density. The higher plasma current data on JET-ILW has much higher density than at low plasma current in part due to gas fuelling, to control W accumulation, and therefore much lower temperature. It is likely that this accounts for the disappointing confinement if there is insufficient power to maintain a hot pedestal. This behaviour is clear from figure 8 where $H_{98(y,2)}$ is seen to strongly increased for $\beta_{N,th}$ above ~1.6 with a fast rise of the pedestal pressure, mainly due to the increase of the pedestal temperature. Higher $\beta_{N,th}$ is achieved at high additional input power. It is then important to understand if proximity to the power necessary to transit from L-mode to H-mode ($P_{L\text{-}H}$) is a factor on the confinement degradation at high plasma current with the ILW. To estimate the additional power necessary to obtain good confinement behaviour at high plasma current, $P_{L\text{-}H}$ was determined at low and high plasma currents at fixed q_{95}, with varying B_T and average density.

Figure 9 (a) shows the measured power threshold dependence on B_T and average density according to scaling in reference [21] as well as the power threshold predicted by the scaling (P_{LH08}). The comparison shows that at lower plasma currents the scaling law overestimates the power necessary for the L-H transition whilst at higher plasma current underestimates it. A positive dependence with I_p is found $P_{L-H} \propto I_p^{0.33}$ (figure 9b). Experiments dedicated to the P_{L-H}, where the plasma current was varied at constant BT also show a dependence of P_{L-H} at low plasma currents [22]. The conclusion is that at this higher current and high density, additional input power is necessary to achieve good energy confinement.

4. SUMMARY AND PROSPECTS

Ultimately, the aim of experiments at JET is to achieve a scenario that has high fusion power when operating in DT within the technical constraints. Cooling of the divertor target by extrinsic impurities, the use of He3 for ICRH heating, ELM mitigation either by pellet fuelling/pacing or by using radial magnetic perturbations coils (RMPs) are all tools to optimise the final plasma performance. This paper focused on how to develop robust scenarios to mitigate and avoid disruptions, on W

accumulation and exhaust control and its impact on plasma confinement and performance. At JET, the RMPs current is not high enough to have any impact on the plasma edge at high plasma current and the pellet system is not reliable enough and it is being modified to improve reliability for 2015.

A small radial sweeping also has no impact on confinement although a larger radial sweep moving the strike points further away from the pump throat does have a small impact on plasma confinement. The discharge length is then defined by a compromise between exhaust and plasma performance. The JETILW is challenging, due to W sputtering and its impact on confinement while mitigating W accumulation, and because of its constraints on plasma operation, namely the limitations on tile surface temperature that imposes limitation on the discharge length. Nevertheless, since the last IAEA, progress on how to operate high power, high stored energy plasmas at JET has been made with the potential for further improvements. The development of two scenarios that aim at high performance in DT has started. Results from other experiments that help to improve confinement as well as to operate safely in a metallic machine are being integrated in the scenarios. The present results predict a fusion power in DT of 15MW with $H_{98(y,2)}$~1.3 at I_p=3.5MA. However, this scenario is not stationary due to MHD and work is on-going to extend the pulse duration whilst maintaining good confinement. The estimated DT-equivalent for the baseline scenario is ~8MW stationary at $H_{98(y,2)}$~0.85 or 13.2MW at $HH_{98(y,2)}$~1 with Z_{eff}~1.12 at 4.5MA/3.83T with an additional heating power of 34MW. These estimates are based on a JET-ILW pulse at 3.5MA for the baseline scenario maintaining the same conditions, including the mass scaling of confinement and compensating for the over prediction of DD neutrons. Work on the improvement of confinement on this scenario is ongoing. As discussed in the paper, common to all scenarios are the limitations imposed by the divertor, however, specific physics issues for each scenario will continue to be addressed in order to prepare for the DT campaign at JET.

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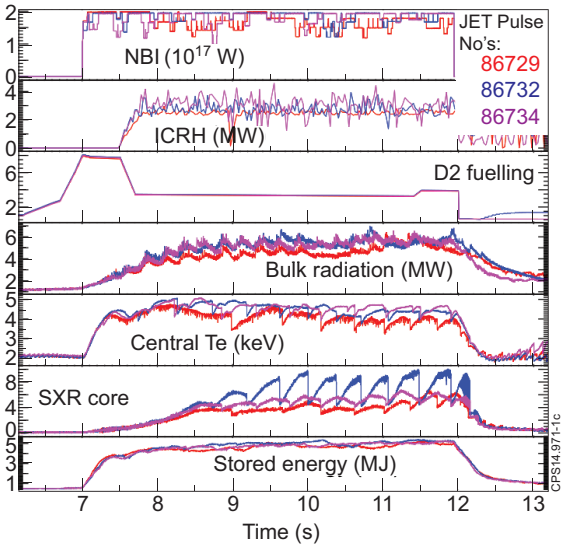

Figure 1: Comparison of ICRH heating on W accumulation control with H (red) and He^3 minority (blue – 2-3% He^3, magenta – 7.6% He^3).

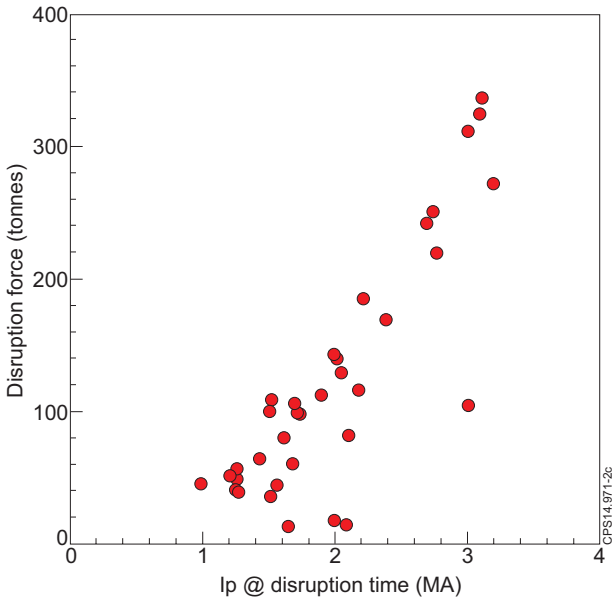

Figure 2: Mitigated disruption forces by MGI.

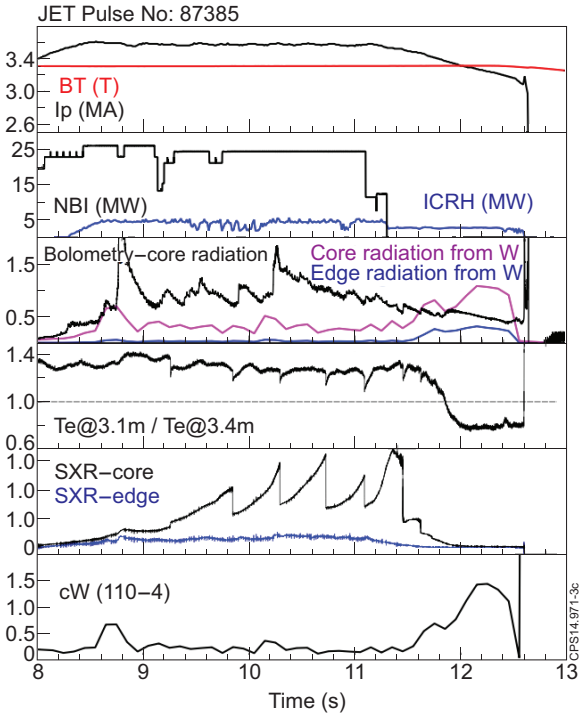

Figure 3: Termination for a low beta plasma with low W accumulation.

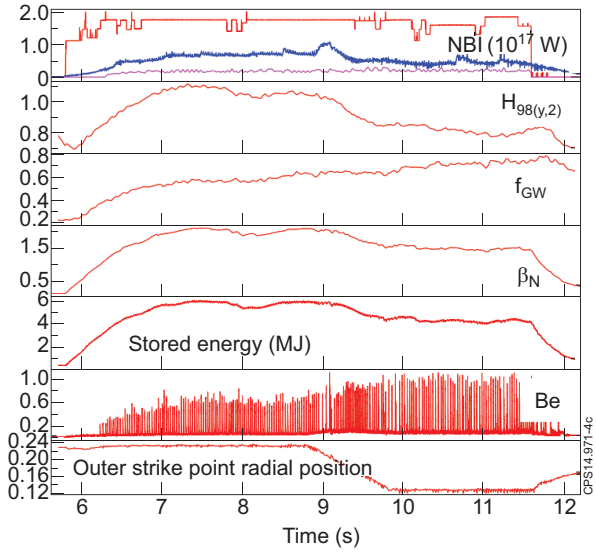

Figure 4: Impact of the outer strike radial position distance to the pump throat (higher R).

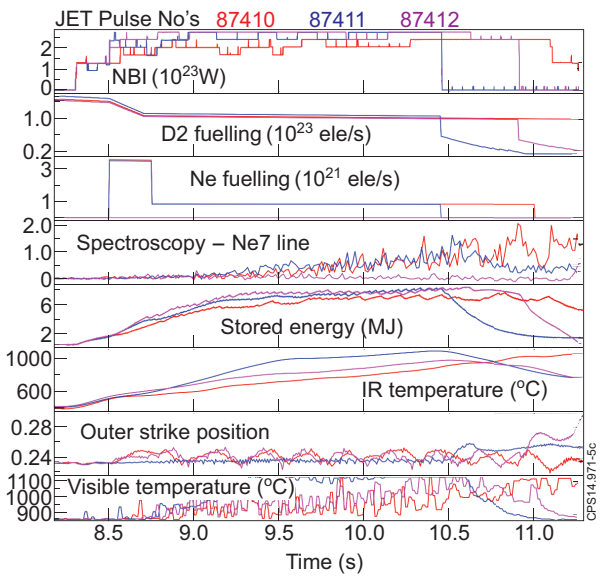

Figure 5: Comparison of sweeping and Ne injection efficiency for surface divertor temperature cooling.

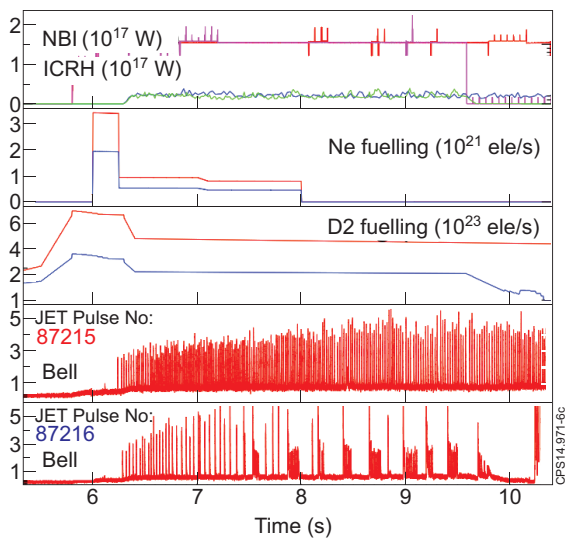

Figure 6: Impact of D_2/Ne on ELMy H-mode behaviour.

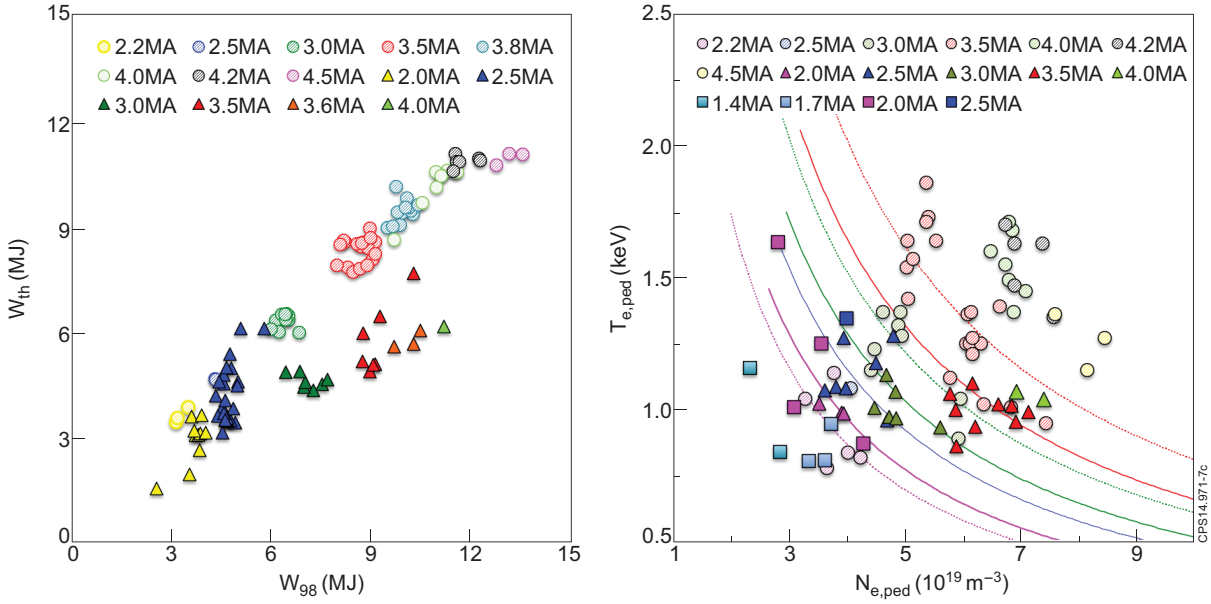

Figure 7: (a) Comparison of thermal energy evolution with IPB98(y,2) scaling and (b) n_e-T_e diagram for high performance scenarios for JET-C (circles) and ILW (baseline – triangles and hyrid squares). Pressure curves are also shown for JET-ILW (line) and JET-C (dashed line).

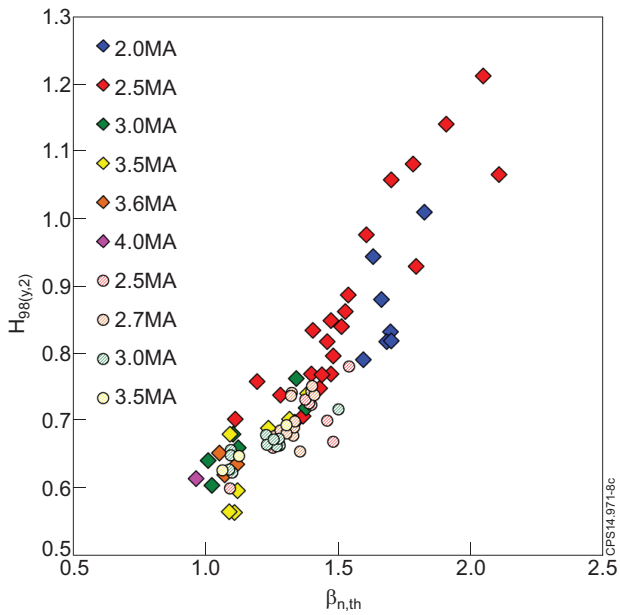

Figure 8: JET-ILW progress from 2012 (circles) to 2014 (diamonds) shown by the confinement and beta achieved for different plasma currents.

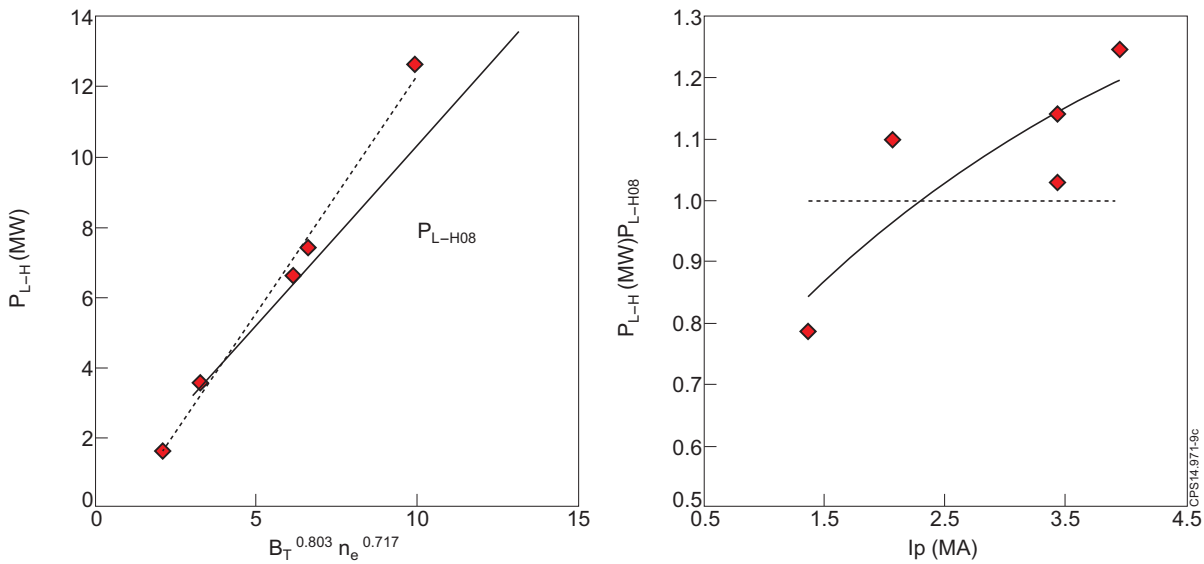

Figure 9: (a) $P_{L\text{-}H}$ measured at I_p = 1.4, 2.1, 3.5 and 4.0MA (red diamonds) compared with $P_{L\text{-}H}$ scaling [21]. (b) $P_{L\text{-}H}$ dependence on plasma current as calculated for this experiment.

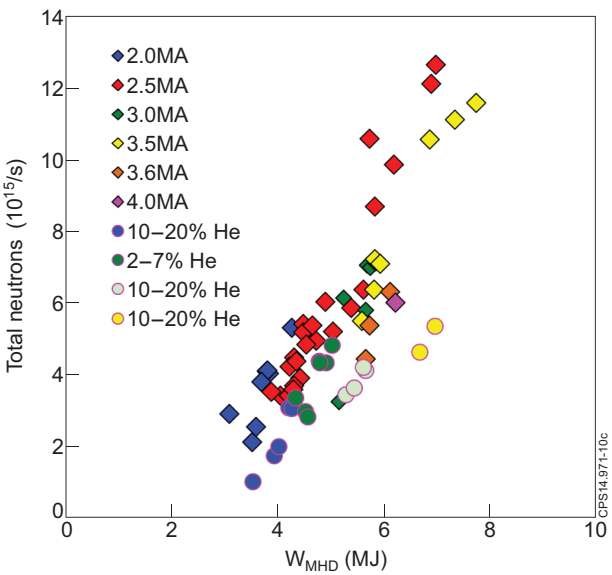

Figure 10: Total neutron rate for the discharges used for the development to high plasma current and discharges with He injection.