Integrated Scenario with Type-III ELMy H-mode Edge: Extrapolation to ITER

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One of the most severe problems for fusion reactors is the power load on the plasma facing components. The challenge is to develop operation scenarios, which combine sufficient energy confinement with benign heat loads to the plasma facing components. The radiative type-III ELMy H-mode seems a possible solution for such an integrated ITER scenario. Most notably the transient heat loads due to type-III ELMs are acceptable with even the most stringent boundary conditions. For instance, on JET the transient heat loads due to type-III ELMs onto the outer divertor target were reduced to 2 kJ per square meter. Scaled to ITER, type-III ELMy H-modes are expected to have a power load of approximately 0.3 MJ per square meter transiently. This was achieved in experiments carried out with nitrogen seeding to mitigate the transient and steady state heat flux to the divertor. Typically the confinement is reduced by ~ 8-20% compared to the type-I ELMy H-mode base scenario. However, increasing the plasma current to 17 MA on ITER and hence reducing the edge safety factor to 2.6, would allow Q=10 operation at a reduced confinement enhancement factor. This operation scenario was demonstrated at JET up to plasma currents of 3.25 MA. At the highest plasma current the effective charge Z_{eff} can be as low as 1.4, mainly due to the increased absolute density and reduced carbon erosion. A large database of highly radiative type-III ELMy H-modes on JET is used for extrapolations to ITER. The data set shows no apparent dependence of the confinement enhancement factor on collisionality. The scaling of the confinement time with respect to the ion gyro radius is close to gyro-Bohm scaling. The 'hybrid' regime, designed for high beta stationary scenarios, has been extended recently at JET to the type III-ELMy H-mode operation by nitrogen seeding (at a plasma current of 1.7 MA) up to a normalized pressure (beta) of 2.6. Similar to the standard ELMy H-mode the confinement enhancement factor is reduced by about 20%. The 'hybrid' type III-ELMy H-mode scenario shows improved edge plasma condition without significant modification of the q-profile (stabilized near unity in the plasma core in order to reduce the sawteeth activity), indicating it is compatible with high beta operation (optimized for current drive sources). Extrapolations to ITER are done with an integrated core/edge model.

Introduction

One of the most severe problems for fusion reactors is the power load on the plasma facing components. Technically only loads of less than 10 MW/m^2 in steady state and less than 0.5 MJ/m^2 [1] during transients, caused by so-called Edge Localized Modes (ELMs), are acceptable. This effectively means that the unmitigated type-I ELMy H-mode is not acceptable for ITER. The challenge is to develop alternative scenarios, which combine sufficient energy confinement to achieve fusion power amplification factors of Q=10, with benign heat loads to the plasma facing components. The radiative type-III ELMy H-mode seems a possible solution for such an integrated ITER scenario. Most notably the transient heat loads due to type-III ELMs are acceptable with even the most stringent boundary conditions. This was achieved in experiments carried out with nitrogen seeding to mitigate the transient and steady state heat flux to the divertor. Typically the confinement is reduced by ~ 8-20% compared to the type-I ELMy H-mode baseline scenario. The reduction in stored energy can be regained by either (a) increasing the plasma current or (b) increasing the plasma core confinement. Both routes have been investigated at JET with the standard ELMy H-mode and the so-called Hybrid scenario.

Standard highly radiative nitrogen seeded ELMy H-mode with type-III ELMs

Increasing the plasma current to I_p=17 MA on ITER and hence reducing the edge safety factor q_{95} to 2.6, would allow Q=10 operation at a confinement enhancement factor of $H_{98(y,2)}$=0.75 at a high density of $n_e/n^{GW}=N^{GW}=1$. The target values for this ITER operation scenario are $\nu^*=0.042$, $\rho^*=0.0015$, $N^{GW}=1$, $Z_{eff}<1.7$, $n_e=13.4\text{x}10^{19}\ m^{-3}$, W_{th}=325 MJ, $W_{ped}=0.35\text{x}W_{th}$, T_{ped}=2 keV and f_{rad}=0.75. This operation scenario was demonstrated at JET in a standard inductive scenario. To obtain these high densities high triangularity plasma configurations had to be chosen. All normalized parameters, including confinement $H_{98(y,2)}$, density N^{GW}, normalized pressure β_N, radiative power fraction f_{rad}, plasma effective charge Z_{eff} were met simultaneously [2]. In recent JET campaigns this plasma regime has been extended to plasma currents of 3.25 MA. In those high current discharges central line averaged densities of up 12 x $10^{19}\ m^{-3}$ were reached.

Hybrid scenario with nitrogen seeding and type-III ELMs

An integrated hybrid type III ELM regime with β_N = 2.6 (P_{NBI} ~ 20-22 MW) and $H_{98(y,2)}$~0.83 has been successfully developed on JET with nitrogen seeding. The target plasma scenario is a hybrid H-mode [3] (defined here as an optimized scenario for high β_N operation with moderate MHD activity) with type-I ELMs (#68505), where pedestal plasma temperature is T_{ped} ~ 1 keV, I_p = 1.7 MA, toroidal magnetic field B_t = 1.7 T, n_e ~ 5 x $10^{19}\ m^{-3}$ (~ 70% of the Greenwald density), q_{95} ~ 3.2 in which injected neutral beam power (NBI) is feed-back controlled to ~14-16 MW to achieve a β_N of 3. The thermal confinement enhancement factor achieved in the target plasma scenario is $H_{98(y,2)}$ ~ 1.05 and the plasma effective charge is Z_{eff} ~ 1.8. A high triangularity magnetic configuration (δ = 0.44) is used. Lower hybrid heating is used during the plasma current ramp up (for a duration of ~ 3s) to delay the plasma current profile penetration with the aim of producing a broad q-profile when the main heating is applied. This is followed by an intermediate β_N = 2 phase (for a duration of ~ 3s) for stabilization of the q-profile close to 1 in order to minimize the impact of sawtooth on stability. The β_N request is then increased and kept constant during 4 seconds. During this phase, a pre-set injection of deuterium is applied: β_N ~ 3 has been obtained with low deuterium fuelling (#68505: $\Phi_D = 0.6\times10^{22}$ electrons per second) and β_N = 2.6 with high deuterium fuelling and density close to the Greenwald density n_e ~ 0.95 x n_{GW} (#68515: $\Phi_D = 5\times10^{22}$ el/s). Nitrogen injection is applied in addition to deuterium fuelling during the first three seconds of the high β_N plateau (when β_N = 2.6). Deuterium is injected in the bottom of the divertor near the outer strike point on the Low Field Side (LFS) while nitrogen is injected into the private-flux region from the horizontal target plate located on the High Field Side (HFS). The maximum radiated power fraction achieved with deuterium fuelling alone (with mainly D and C radiators) is P_{rad}/P_{heat}~37% with density close to the Greenwald limit $n_e \sim n^{GW}$ (#68739). Using deuterium plus nitrogen fuelling enables to increase the radiative fraction (with mainly D, C and N radiators), ~50% has been achieved during the experiment. The type III ELM regime is achieved here when P_{rad}/P_{heat} > 40 % and the pedestal ion temperature T_{ped}

is reduced to values below 750 eV. The degradation of global confinement associated with the type-III ELM regime is about 10% compared to the reference hybrid high D-fuelling discharge (#68515), which uses the same D-fuelling: Φ_D=5 x10^{22} el/s.

Power load to the plasma facing components

It has been found that the power load to the divertor can be reduced significantly in the nitrogen seeded type-III ELMy H-mode. In steady state radiative power fraction of 97% were already achieved [2]. Furthermore it should be noted that the transient heat loads due to type-III ELMs can be reduced significantly in type-III ELMs [4,5]. For instance, on JET the transient heat loads due to type-III ELMs onto the outer divertor target were reduced to 2 kJ/m^2 [4]. With the new infrared diagnostic capabilities [6] better measurements with improved time resolution were done. For the highly radiative nitrogen seeded plasmas the temperature excursions in the outer divertor due to the type-III ELMs is only in the range of 10 degrees C. This corresponds to a power flux density of about 1 MW/m^2. At the inner divertor it is typically half of that value. No difference between the standard type-III ELMy H-mode and the hybrid type-III ELMy H-mode was observed. The divertor heat load due to the type-III ELMs normalized to the total stored energy versus the core collisionality is shown in figure 1. The data set shows no apparent dependence of the ELM energy deposition in the divertor as a function of collisionality. If at all the ELM energy loss is slightly decreasing with decreasing collisionality. The absolute value is about 0.1%. Taking a constant $\Delta W_{div}/W$ of 0.1% for type-III ELMs in a radiating scenario the ITER divertor load would be about 0.3 MJ. This would approximately translate into 0.1 MJ/m^2. However, if a collisionality dependence of $\Delta W_{div}/W$ similar to type-I ELMs is assumed, then the energy load to the divertor could be a factor of 3 higher, leading to a predicted energy load of 0.3 MJ/m^2. It appears that the ELM rise time for type-III ELMs is much slower than for type-I ELMs. Details of the temporal evolution and spatial energy distribution could influence the predictions and could lead most likely to an acceptable energy load to the divertor higher than the currently adopted limit of 0.5 MJ/m^2. Even more difficult to predict is the ELM frequency of the radiative type-III ELMs. At JET the ELM frequencies vary between 150 Hz and 1 kHz.

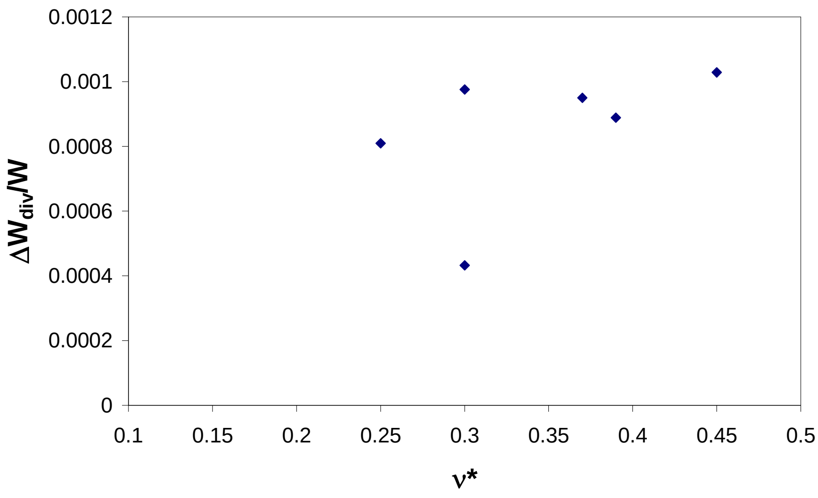

Figure 1: $\Delta W_{div}/W$ as a function of collisionality as derived from IR thermography for 2.5MA/2.0T type-III ELMy H-modes

Confinement scaling

The main drawback of the type-III ELMy H-mode is it's reduced confinement, when compared to the type-I ELMy H-mode. The JET steady-state database contains something like 576 type-III ELMy H-modes and 672 type-I ELMy H-modes. The fit gives a $H_{98(y,2)}$ of 0.95 for the type-I ELMy H-modes and a $H_{98(y,2)}$ of 0.87 for the type-III ELMy H-modes [7]. The parametric dependence in the scaling is otherwise the same for type-I ELMy H-modes and type-III ELMy H-modes. However, for strongly radiating type-III ELMy H-modes the confinement is slightly lower. To investigate this further, a large database of highly radiative

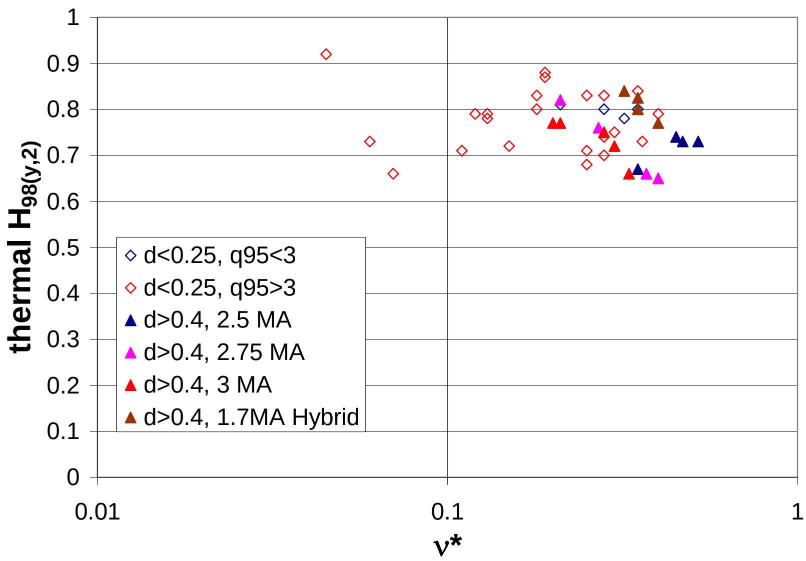

Figure 2: $H_{98(y,2)}$ as a function of collisionality for radiative type-III ELMy H-modes at JET

type-III ELMy H-modes on JET has been set up, including discharges in low and high triangularity configurations. Figure 2 shows the confinement enhancement factor H98(y,2) as function of the collisionality.
The data set shows no apparent dependence of the confinement enhancement factor on collisionality. If at all the confinement is slightly increasing with reduced collisionality.

In figure 3 the $H_{98(y,2)}$ is shown versus the normalized gyro radius. The operational domain covered at JET does extend from 0.9 MA/ 1 T AUG, CDH-mode identity pulses up to 3 MA with high and low q_{95}, as well as 2.5 MA / 3.45 T low ρ* pulses with heating powers of up to 33 MW. Within one plasma current and configuration the $H_{98(y,2)}$ does decrease with decreasing ρ*. This is an effect of the increased gas fuelling in those experiments, which leads to lower pedestal as well as core temperatures. However, all those lines are parallel to each other. Comparing the data at high ρ* with the 2.5MA/3.45T pulses leads to the conclusion that the confinement scales like $\rho_*^{-2.5}$. However this could be due to a different β dependence in the $H_{98(y,2)}$ scaling [8]. Neglecting this would lead to a scaling, which is closer to gyro-Bohm scaling: $B\tau_{E-gyro-Bohm} \propto \rho_*^{-3.0}$. It needs to be proven with more data at low ρ* that this is true.

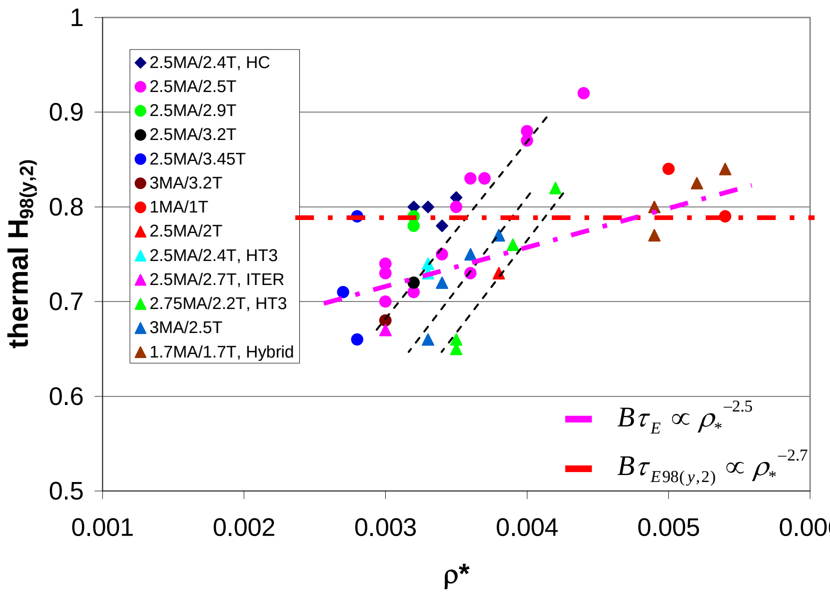

Figure 3: $H_{98(y,2)}$ as a function of normalized gyro radius for radiative type-III ELMy H-modes at JET

Edge operational space

By lowering the collisionality the pedestal temperature was increased from 0.4 keV to about 0.9 keV in the high triangularity discharges (see figure 4). This has to be extrapolated to the 17 MA ITER Q=10 scenario at N^{GW}=1, for which a pedestal temperature of 2-2.5 keV is necessary [9, 10]. Figure 4 shows a large gap between the JET pedestal temperature data and

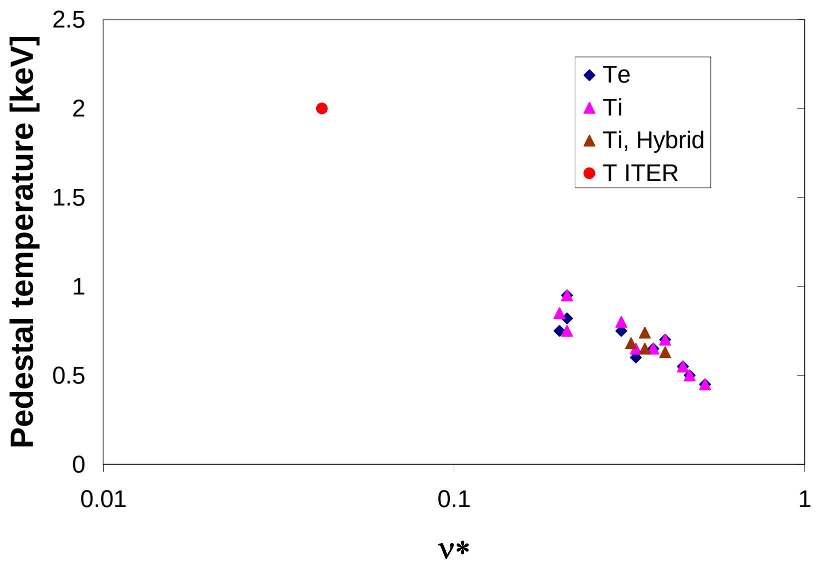

Figure 4: Pedestal temperature as a function of collisionality for radiative type-III ELMy H-modes at JET

the ITER pedestal temperature exists. The model also predicts that the gap between type-I ELM pedestal temperatures and type-III ELM temperatures is closing in at higher densities. However, the model does not reflect the q_{95} dependence of the experiments [11]. Hence some uncertainties in the prediction of the type-I to type-III ELMy H-mode threshold remain.

Plasma pollution

The plasma pollution (Z_{eff}) depends strongly on the absolute density in those highly radiating plasmas. In recent JET campaigns the standard radiating type-III ELMy H-mode has been extended to higher plasma current (3.25 MA) and therefore high absolute density. At those high densities the Z_{eff} is strongly reduced. The effective charge Z_{eff} was reduced from ~ 2.2 to values below 1.5, mainly due to the increased absolute density and reduced carbon erosion. In those highly radiative discharges (f_{rad}>0.75) nitrogen having replaced carbon, is the main radiator and the dominant impurity in the plasma. A comparison of the carbon erosion in the divertor and the main chamber wall, revealed that the main chamber wall erosion is much lower in type-III ELMy H-modes, when compared to type-I ELMy H-modes [10]. Naturally, for the type-III ELMy hybrid mode the Z_{eff} is higher since the density is lower and the heating power is much higher. At a density of n_e=7 x 10^{19} m^{-3} the Z_{eff} is about 3.

On basis of an enlarged database a new Z_{eff} scaling has been developed [12]: $Z_{eff} = 1 + 40 P_{rad} Z^{0.12} \tau_E S^{-0.94} n_e^{-1.5} a_{\min}^{-1} R^{-1}$, with S being the plasma surface, a_{min} the minor radius and R the major radius. For the high density 17 MA scenario with a fusion power of 400 MW and a $H_{98(y,2)}$=0.75 a Z_{eff} of 1.9 is predicted, excluding any contribution from Helium. This is a little above the assumptions made for ITER, which include Helium. However, details of the radial impurity transport and profile effects in the temperature and density profiles are not taken into account.

MHD stability

All ELMy H-modes at JET are marginal unstable to the development of neoclassical tearing modes (NTMs) [13]. The radiative scenarios are particularly prone to NTMs, because of its dependence of the marginal beta β_{Nmarg} on the normalized poloidal ion gyroradius $\rho^*_{p,i} = \rho_{p,i}/a$ [14], which is low in the radiative scenarios with the cooled pedestal. Seed islands produced by large sawtooth crashes can trigger the NTMs, when β_N is larger than the critical marginal β_{Nmarg} [13]. The disturbance and hence the seed island is particular large in sawtooth crashes where a large volume is affected, hence large sawtooth inversion radius. Operation at q_{95}=2.6 has the disadvantage of having a large sawtooth inversion radius (see figure 5), which potentially leads to larger sawtooth crashes. A scenario had to be developed to avoid triggering the NTMs (m/n=3/2) in the transient period of the ramp up in the radiative scenario, where the sawtooth period is the longest. Setting q_{95}=3.6 at the beginning of the heating period and ramping down to q_{95}=2.6, once regular sawtooth activity is obtained is the major recipe to avoid NTMs. During the flat top period of the highly radiative scenario, sawtooth activity is benign. In particular at the high plasma densities, which are an integral part of the highly radiative type-III ELMy H-mode, the relative sawtooth amplitude is small (see figure 6).

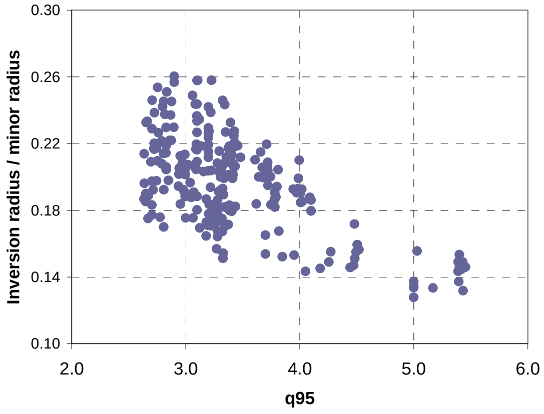

Figure 5: Sawtooth inversion radius normalized to minor radius versus q_{95} for JET NBI heated plasmas.

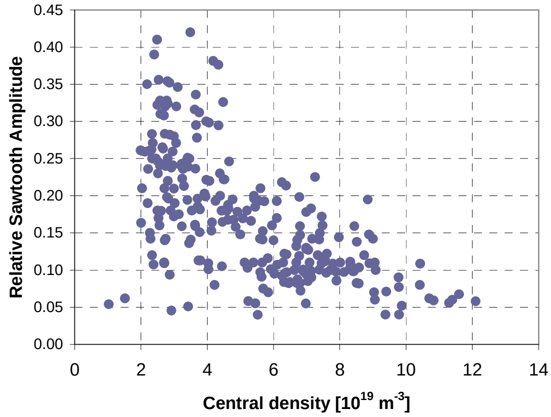

Figure 6: Relative sawtooth amplitude versus central density for JET NBI heated plasmas.

The MHD activity observed during the hybrid type-III ELM operation is characterized by mild n=1 sawteeth precursors present during the high β_N and n=3 mode presents near the q = 4/3 surface. The more deleterious 2/1 and 3/2 NTM are not present during the hybrid type-III ELM case and nor in the reference type-I case. The core q-profile of the integrated hybrid type III ELM is very similar to the q-profile of the reference hybrid type I ELM scenario (figure 2), indicating that type III ELM operation obtained with N-injection is compatible with high β_N operation with minimized impact of sawteeth on the stability.

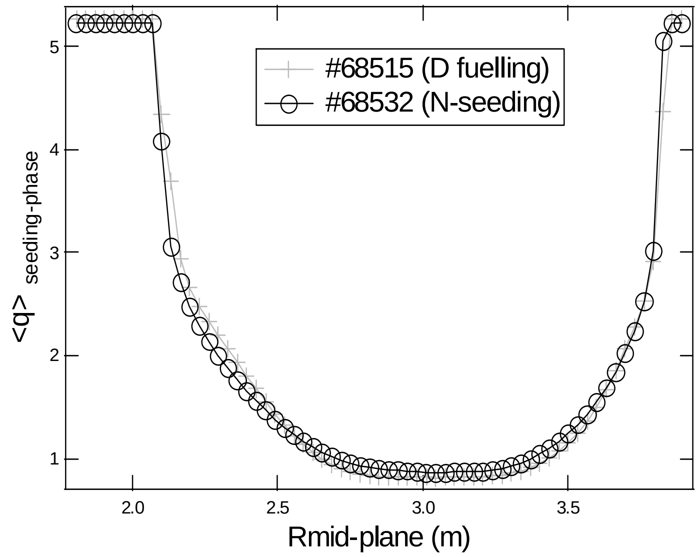

Figure 7: Time average of q-profiles measured by the MSE system during the seeded phase t=[8-11s] for the pulse with D fuelling with type-I ELMs (crosses) and D+N fuelling with type-III ELMs (circles).

Extrapolation to ITER with the integrated model COREDIV

The nitrogen seeded high triangularity JET discharges have been modelled with COREDIV [15]. COREDIV is an integrated model solving self-consistently the 1D energy and particle transport of plasma and impurities in the core region and 2D multifluid transport in the SOL [16]. The energy confinement is scaled with the empirical $H_{98(y,2)}$ scaling. The target erosion by nitrogen is not included. The carbon target erosion is dominated by deuterium and self sputtering. The chemical erosion yield is calculated according to the flux dependence given in

[17]. No main chamber erosion was included in the calculations. However, the experimental data (main plasma profiles in the core, the radiated power and the plasma pollution) were reconstructed satisfactory.

On the basis of this benchmarking to JET experiments predictions for ITER were done. In the ITER case neon was seeded as radiating impurity. The target was again a carbon target. Figure 5 shows the results of the COREDIV simulations for the 15 MA standard ITER scenario for a set of densities and confinement enhancement factors. The results indicate, with reasonable accuracy, that this plasma scenario can achieve a power amplification factor Q in excess of 6 at 15 MA (q_{95} = 3.0) with auxiliary heating powers of 40 MW. Higher heating power will have a detrimental effect on the fusion amplification factor.

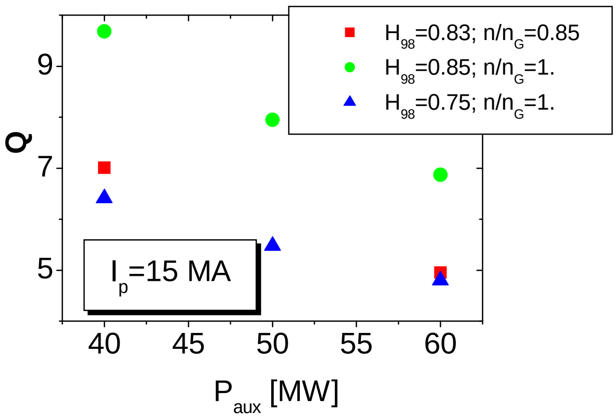

Figure 8: Fusion amplification factor Q versus the auxiliary heating power for the ITER 15MA scenario for radiative scenarios

For 17 MA (q_{95} = 2.6) the extrapolations with the code show the compatibility of those strongly radiating type-III ELMy H-modes with a power amplification in excess of 10 (see figure 8). In both cases, 15 MA and 17 MA, the plasma core pollution is below Z_{eff}=1.5 (see figure 9 and 10). This does not include any contribution from the Be-wall. Assuming a core concentration of 2% of Be in the plasma core would then increase the Z_{eff} to about 1.7, consistent with former ITER predictions [18].

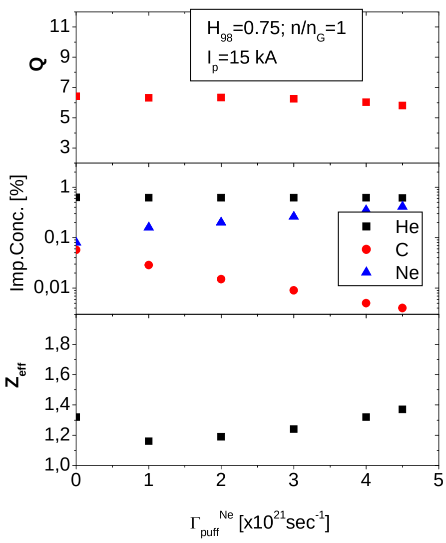

Figure 9: Fusion amplification factor Q, impurity concentration in the core plasma and Z_{eff} versus the neon gas fuelling rate for ITER 15MA scenario

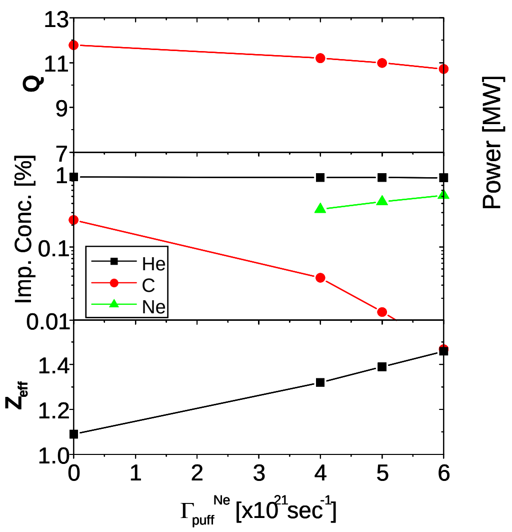

Figure 10: Fusion amplification factor Q, impurity concentration in the core plasma and Z_{eff} versus the neon gas fuelling rate for ITER 17MA scenario

Conclusion

A large data base on radiative type-III ELMy H-mode at JET does allow a reasonable extrapolation to ITER. Within the error bars of the data and based on the simplicity of the extrapolation models the extrapolation to ITER does allow the following statements: (a) the confinement (transport) should be sufficient to reach Q=10 at 17MA depending on point d; (b) the steady state heat load will be reduced to acceptable values; (c) the transient heat load should be acceptable even with respect to the most stringent limits; (d) the plasma pollution and hence the plasma core dilution could be slightly to high, leading to some reduction in plasma performance; (e) the accessibility of the type-III ELMy regime at higher pedestal temperatures needs to be still proven; operation at low q_{95} seems to be possible and reliable.

If the confinement time of future ITER hybrid discharges (presently foreseen at low plasma current I_p~14MA) is high enough to allow type-III ELMy operation with acceptable fusion performance (Q>5), then the experimental procedure described here can be envisaged to control the edge plasma conditions and get sustainable heat load (compatible with the ITER walls) without modifying the core q-profile, and thus the high β_N capability of the hybrid scenario (optimized for current drive sources and non-inductive current bootstrap). The relatively high impurity content and the extrapolation to ITER remain important issues to demonstrate the viability of the hybrid type-III ELM scenario as an integrated scenario for ITER. The use of the real-time control, maximization of confinement as observed in ASDEX Upgrade [19] and impurity decontamination techniques will be essential to improve the performances and the reliability of the scenario.

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