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ABSTRACT

A review of MHD limits to tokamak operation in terms of current, density and pressure is given. Although the current and density limits in a toka-mak usually lead to disruptive termination of the discharge, it is argued that these can be avoided by staying away from the respective limits. This is especially true since operation close to these limits is not really desirable, due to the decreased confinement at very high density and the high disruptivity at low q. On the other hand, the limit to plasma pressure set by Neoclassical Tearing Modes and Resistive Wall Modes is too low to guarantee economic operation of future fusion reactors. Therefore, active control of these two in-stabilities is now being studied. Noticeable progress has been made by NTM stabilisation with ECCD. Avoidance of NTMs and RWMs by tailoring sawteeth and spinning the plasma, shows promising results. Also, experiments on direct RWM stabilisation by active coils are showing their first encouraging results.

1. INTRODUCTION

The plasma parameters achievable in toroidal magnetic confinement devices used for Nuclear Fusion research are limited by large scale Magnetohydrodynamic (MHD) instabilities. In the tokamak configuration, for a given toroidal magnetic field B_t, the maximum plasma current I_p, particle density n and plasma pressure p are each limited by the occurrence of different MHD modes. This limitation is important because the energy confinement time τ_E scales linearly with I_p and in order to ignite a fusion plasma, a certain value of $n\tau_E$ has to be reached at a temperature of 10-20keV. At these temperatures, the fusion power scales with p^2, so all three parameters are crucial for the performance of a future fusion reactor [1].

This paper discusses these limitations and explains our current physics un-derstanding of the MHD processes involved. The maximum achievable I_p is limited by edge current gradient driven tearing and kink modes. While the ultimate limit for I_p is set by the condition $q(a)$ =2, where q is the safety factor ($q = 2\pi r^2 B_t / (\mu_0 R_0 I_p)$ in cylindical approximation), tokamak discharges become more prone to disruptions in the region $q(a) < 3$, so that present designs for next step devices usually plan operation at $q \geq 3$. The occurrence of a radiative instability at the plasma edge (MARFE) [2] limits n, which, via contraction of the current profile, can also ultimately lead to the occurrence of current gradient driven tearing modes resulting in a disruptive termination of the tokamak discharge [3]. The main limitation to p in a tokamak with conventional current profile is the Neoclassical Tearing Mode (NTM) [4], whilst with advanced current profiles for which the NTM is stable, the ultimate limit is set by a pressure and current driven kink mode which, in the presence of a conducting wall, becomes a Resistive Wall Mode (RWM) [5].

Based on this understanding, the paper discusses strategies to avoid or actively control these instabilities. The main emphasis is on the areas in which recent progress in active control of the pressure limiting instabilities has been achieved, namely NTM suppression by Electron Cyclotron Current Drive (ECCD) and RWM suppression by rotation and error field control. Alternative schemes will be discussed and compared concerning their relevance to future reactor-scale tokamak devices such as the International Tokamak Experimental Reactor ITER. We will conclude with a discussion of open issues and strategies to resolve these.

2. THE DENSITY LIMIT

In the classical density limit experiment, the density limit in a tokamak is infered by continuous strong gas puff fuelling which finally leads to a saturation of density increase with a subsequent energy collapse and a disruptive termination of the discharge. Empirically, it has been found that this limit is roughly proportional to the current density and the critical density is known as the Greenwald density $n_{GW} = I_p = \pi a^2$ where n is in 10^{20} m^3, I_p is in MA and the minor plasma radius a in m [6].

More refined analysis of this kind of density limit showed that the saturation of density increase is linked to a power balance problem at the plasma edge. If the edge cools to a sufficiently low temperature of 50-100eV, a radiative instability can occur due to the effect that a small concentration of impurities changes the plasma radiation chracteristics in such a way that, with decreasing temperature, an increasing radiative loss occurs. In this thermodynamically unstable situation, a toroidally symmetric, but poloidally localised zone of very cold plasma (MARFE) can occur. Note that in the presence of a MARFE, the pressure on the flux sur-faces is still constant, as required by ideal MHD, but density and temperature show a great variation (down top 10 eV locally in the MARFE) because at these low temperatures, parallel heat conductivity is no longer able to equilibrate parallel temperature gradients [7]. With such a low temperature at the plasma edge, the electrical conductivity also decreases drastically and the current profile con-tracts. This contraction leads to a steepening of the current density gradient at peripheral rational surfaces, which in turn can destabilise classical tearing modes. In tokamaks, usually the (m = 2, n = 1) mode (where m and n denote the poloidal and toroidal mode number) is dominant before disruptions, but also other islands of low toroidal mode number are observed. The occurrence of a number of island chains of different helicity ultimately leads to stochastisation, which in turn leads to a rapid energy loss [8], [9]. At the resulting low temperatures of only several 10eV, the tokamak transformer can no longer sustain the current and the discharge disrupts.

Disruptions can lead to excessive forces and thermal loads on the tokamak and are therefore a main concern for future tokamak devices. Mitigation techniques such as the injection of impurity pellets [10] or gas jets [11] have been developed to soften these consequences. However, in the last years it has also been realised that the maximum density at which a tokamak reactor will operate is probably not determined by the classical density limit, but more by a confinement reduction that sets in when n is close to n_{GW}. For example, density limit disruptions never occur in H-mode, they are always preceded by an H-L transition at high density. Since H-mode is probably mandatory in future devices (at least in conventional scenarii), this is a practical limitation to tokamak operation. Fig.1(a) shows data from the ASDEX Upgrade tokamak where the H-mode operational domain has been mapped out in terms of heating power close to n_{GW} [12]. It is clearly seen that the power needed to stay in H-mode increases more strongly than the conventional H-mode threshold scaling when n approaches n_{GW}. In fact, the density at which the H-L back transition occurs roughly scales with the Greenwald density. Increasing the plasma shaping (mainly the triangularity of the poloidal cross-section) is found to positively influence this limit. With sufficient shaping, good H-mode confinement can be reached even at n_{GW} as is shown in Fig.1(b) [13].

From this discussion we conclude that the practical density limit for tokamaks is not set by the disruptive MHD limit, but rather by the onset of confinement degradation. It is thus not primarily an issue of control of MHD activity. However, since the confinement degradation sets in close to n_{GW}, operation is envisaged close to the disruptive limit and thus mitigation techniques are required in the case of an abnormal event which may lead to a disruption.

3. THE β-LIMIT IN CONVENTIONAL SCENARII: NEOCLASSICAL TEARING MODES

The ideal β-limit of tokamak discharges is usually set by the onset of kink or ballooning modes. This critical β-value scales as β_{crit}~I_p = (aB) [14], which leads to the introduction of the so-called normalised beta $\beta_N/\beta/(I_p/(aB))$. In con-ventional scenarii with positive magnetic shear and $q(a)\approx 3\text{-}4$ the maximum achievable β_N is in th range 2.8-5, where 2.8 refers to a cylindrical cross-section, whereas the upper bound is reached with extreme shaping of the plasma cross section. This value would be sufficient for economic operation of a future fusion reactor. However, in long pulse discharges of low collisionality, the β-limit scenarii is usually set by the occurrence of resistive MHD insta-4 bilities, namely Neoclasscal Tearing Modes (NTMs) [15], [16]. Thus, a major effort in present fusion research is directed towards understanding NTM physics and devising active control means of NTMs.

3.1 NTM PHYSICS

The basic instability of the NTM is due to the fact that at sufficinet β, the local flattening of the pressure within the island leads to a helical hole in the bootstrap current distribution which, for positive shear, reinforces the island [4]. However, at small island size, a number of effects such as incomplete flattening due to finite parallel heat conductivity [17] or stabilisation due to the polarisation current induced by the mode rotation [18] lead to a stabilisation of the NTM. Thus, the NTM usually requires a finite 'seed' island introduced by another MHD perturbation (e.g. a sawtooth or a fishbone) to be triggered. At larger island size and in the case of a classically tearing stable plasma (i.e. $\Delta' < 0$), the magnetic energy connected with the opening of the island increases and a finite saturated width of the island $W_{sat} \propto \beta_p = \Delta'$ is reached.

Experimentally, it has been found that the NTM onset scales with the normalised poloidal ion gyro radius ρ^*_p [19]. This is demonstrated in Fig.2 for a dataset from JET and ASDEX Upgrade which covers a wide range of ρ^*_p and ν^*_{ii} values [20]. Here, ν^*_{ii} is the ion-ion collisionality normalised to the mode rotation frequency in the plasma frame. This choice of parameter is motivated by the polarisation current theory, which predicts a threshold behaviour for NTM onset in a sense that the onset β_N is drastically reduced when ν_{ii} falls below a numerical value of order unity. It can be seen that by using the proper normalised quantities, a common scaling which is almost linear in ρ^*_p and effectively does not depend on collisionality is found. It must be mentioned that this scaling is consistent with both stabilising mechanisms mentioned above, so that no real discrimination of the model that correctly describes the seed island size can be made on the basis of this experimental result. However, applying this scaling to ITER, which has a much lower ρ^*_p value, predicts that ITER may be prone to NTMs at very low values of β_N. This motivates the development of avoidance or active control techniques for NTMs as discussed in the next sections.

3.2 NTM AVOIDANCE BY SEED ISLAND CONTROL

A possible strategy to avoid NTMs makes use of the fact that even in the non-linearl unstable regime, an NTM needs a finite seed perturbation to grow. Thus, experiments have been conducted to study the effect of different sawtooth sizes on the NTM onset. At JET, Ion Cyclotron Current Drive (ICCD) at the $q = 1$ surface was used to influence the sawtooth period and amplitude [21]. Small and rapid sawteeth permit operation at high β_N without NTM, whereas big isolated sawteeth trigger an NTM at lower β_N.

Since the exact control of the ICCD is not simple, it is also advisable to test if this method can be used with ECCD. Experiments in ASDEX Upgrade have shown that ECCD of a relatively small (order of 10%) fraction of the total heating power is sufficient to stabilise or destablise sawteeth even in the presence of fast particles from the NBI heating [22]. As can be seen in Fig.3, NTM avoidance by sawtooth control is also possible using this scheme. This method is very sensitive to the correct deposition of the ECCD due to the narrow deposition profile which, on the other hand is responsible for the effective local CD with small power requirement. Thus, a ramp of the toroidal field together with a feedback control of b has been applied to compensate for the deposition change due to the change in Shafranov shift when β changes. In future, the deposition must be feedback controlled e.g. by moving the ECRH launching mirror.

3.3 DIRECT SUPPRESSION BY ECCD

A method to directly control NTMs is to inject ECCD at the resonant surface of interest. The current driven here will both change the equilibrium current density, thus changing the stability index Δ', and generate a helical current within the rotating island due to the equilibration of the fast electrons along the field lines of the island structure [23], [24]. This requires precise location of the ECCD at the resonant surface and, in addition, a good radial localisation of the current because current driven outside the island will only slightly contribute to the helical current and may even increase Δ'. Due to this effect, it is also predicted that for the stabilisation of islands with width W below the ECCD deposition width, modulation of the ECCD power in phase with the island O-point is necessary [25]. In present day experiments, this is usually not needed because the deposition width is close to the island width at which the mode is no longer supported by the plasma anyway.

Experiments with DC co-ECCD injection have validated this scheme as a viable option in ASDEX Upgrade [26], DIII-D [27] and JT-60U [28]. The predicted requirements for power and localisation are confirmed by the experiment (the predicted requirement for modulation at small W has so far not been validated). It has also been verified that ECRH is much less effective than ECCD, validating the role of direct current drive and that with ctr-ECCD no direct stabilisation is possible [29]. The difference between helical current and equilibrium current modification is at present only accessible with modelling; typical analysis yields a dominant contribution from the helical current. Conversely, a situation with dominant Δ' change has been observed in COMPASS-D with LHCD [30]. In ASDEX Up-grade and DIII-D, typically, the (3,2) NTM is completely suppressed at ECCD power of roughly 10% of the total heating power. The (2,1) NTM has also been completely suppressed in ASDEX Upgrade [31] and DIII-D [32]; however, the power requirement is higher for this mode than for the (3,2) NTM, consistent with the reduced CD efficiency at larger minor plasma radius. An example of a (3,2) sabiliation experiment from ASDEX Upgrade is shown in Fig.4. Here, B_t is ramped slowly in order to move the ECCD deposition across the resonant surface to ensure correct deposition within the island.

In the discharge shown in Fig.4, it can be seen that β_N is increased above the onset value in the phase when ECCD is on. However, at even higher β_N, the mode comes back even in the presence of ECCD. Analysis of the position of the magnetic island with respect to the ECCD deposition location shows that in this late phase, due to the change in Shafranov shift when β increases, a mismatch between deposition and mode occurs [33]. Thus, the ultimate potential of this method can only be explored by feedback control of the deposition.

An implementation of a feedback control was done on DIII-D, where a so-called 'search and suppress' algorithm stepwise adjusts the major radius in steps of 1cm in ECCD deposition radius, then leaves it constant and detects the change in mode amplitude and, based on this, decides to move the deposition further back, or rest at the present position [27]. With this method, it has been possible to completely suppress (3,2) and (2,1) NTMs even when the initial deposition was not at the optimum position. An example is shown in Fig.5. Alternatively, the toroidal field has been adjusted in a stepwise manner leading to similar results. Feedback control of the poloidal launch angle has been successfully applied in JT-60U to stabilise NTMs. Here, ECE has been used as a sensor: the T_e fluctuations duced by the magnetic island can directly serve as indicator for the mode position and were used on-line to control the mirror angle in successful (3,2) stabilisation experiments in JT-60U [34].

In the experiments discussed above, ECCD was injected into an already existing and saturated NTM. However, theory predicts that the required power to keep an NTM stationary at a certain amplitude has a maximum at certain island width W and then decreases again to smaller island widths. Thus, generally two W solutions can exist for a given ECCD power and it depends on the discharge history whether the larger W-value is obtained (when coming from a saturated mode) or the smaller one (when coming from zero mode amplitude). Since the solution with smaller W is an unstable branch, the mode should effectively decay to zero. There are in fact experimental hints that this behaviour exists: On JT-60U it was shown that ECCD injection into a discharge without NTM prevents the NTM form growing to a larger value after triggering, whereas the same power cannot reduce the mode to equally small amplitude when applied to a saturated mode [35]. However, the mode remains at finite amplitude even with early ECCD, which may be a hint that here, modulation in phase with the isalnd could be beneficial. An example is shown in Fig.6

From these results, it can be concluded that ECCD stabilisation is a serious candidate for an NTM stabilisation scheme in ITER. It should be noted that modelling of the experiments shown here generally gives good agreement, so that extrapolation to ITER can be done. Here, the main problem is the prediction of the NTM stability for ITER, which crucially enters into the power requirements.

4. THE β-LIMIT IN ADVANCED SCENARII: RESISTIVE WALL MODES

We now turn to the stability of advanced scenarii with elevated flat or reversed shear. In these scenarii, NTMs are not expected to be the limiting element because with reversed shear, the missing bootstrap current tends to make the island smaller. In addition, many advanced tokamak scenarii envisage operation with minimum q-value above 2, which would eliminate the two most dangerous NTMs at $q = 1{:}5$ and $q = 2$. Thus, ideal MHD stability may well set the β-limit. However, the combination of a flat or even hollow current profile needed for advanced tokamaks together with a peaked pressure profile leads to a considerably lower ideal stability limit than in conventional scenarii. For example, with reversed shear, external kinks have been found to limit b at values of $\beta_N = 1{:}5$ 2 in ASDEXUpgrade [36]. These values are unacceptably low, especially for the advanced tokamak scenario, which aims at maximising the bootstrap current fraction and therefore needs high β-operation. Thus, control of the external kink is a crucial element of the advanced tokamak development.

4.1 RWM PHYSICS

A possible way of stabilising the etxernal kink is by the introduction of a close fitting conducting shell which, at infinite conductivity, can completely stabilise the external kink if the wall is close enough to the plasma. However, a shell with finite resistivity will only slow down the growth rate of the mode from the ideal (Alfven) time scale to the resistive timescale of the wall, thereby offering the possibility to act on the mode stability with external actuators such as saddle coils which try to cancel the mode perturbation. This branch of the ideal kink is called the Resistive Wall Mode (RWM) [5].

The situation changes if the mode rotates with respect to the wall, because finite rotation effectively separates the two branches. If the slip frequency between wall and mode is well above the inverse wall time constant, the wall can again be considered as ideally conducting and the RWM can be stabilised. The value of the slip frequency is determined by the force balance between wall drag (the eddy currents in a resistive wall tend to slow down the slip frequency) and the force that the plasma, rotating with respect to the wall, exerts on the mode. For an ideal mode, the latter cannot be described by simple MHD, but by several effects, such as sound wave excitation when the flow becomes sonic and tends to produce such a force. Since the wall drag is stronger when the wall is closer to the plasma, the RWM becomes more stable when the wall is moved away from the plasma. Thus, a stability window at b-values above the ideal limit can exist in which the wall is close enough to stabilise the ideal branch, but still far enough from the plasma edge to not slow down the plasma to the wall time constant so that the RWM can grow. Theory predicts that the rotation required to open up this window is of the order of several percent of the Alfven speed.

4.2 RWM AVOIDANCE BY PLASMA ROTATION

The predicted effect of rotational stabilisation of the RWM has been verified in DIII-D [37] and JT-60U [38], where operation above the ideal no-wall β-limit is possible as long as the plasma is rotating fast enough. Figure 7 shows an example where the ideal no-wall limit has been substantially exceeded for a duration of several 10s of the wall time constant (is a reproducible observation in DIII-D. Recent theoretical work suggests that the slowing down is due to an amplification of small helical error fields produced by imperfections in the tokamak coil system [39]. It has been shown that a RWM close to marginal stability can amplify these unavoidable field errors if they have the same helicity which in turn leads to a slowing down of plasma rotation. This hypothesis has actually been verified in the DIII-D tokamak where experiments with an externally applied error field have shown that the external perturbation is strongly amplified when the plasma exceeds the no-wall limit. Recent findings at JET, where similar experiments have been carried out, also find an error field amplification, but here, it seems that no sharp transition is seen when β rises [40]. In DIII-D, it has also been shown that a cancellation of the existing error field by additional saddle coils can to a certain extent avoid the slowing down above the no-wall limit and provide stable operation under steady conditions [41].

Although in present day experiments, the strong rotation induced by the neutral beam heating systems provides a sufficient rotation to open up a stability window above the no-wall limit, predictions of the rotation in future large devices with dominant a-heating power (and high energy neutral beam injection systems which induce less rotation) point out that rotation might not be sufficient to open up a stability window there. Thus, an active feedback control of the RWM may be needed. This is also important in view of the slowing down by error field amplification, which prevents access to the ideal wall limit for finite error fields. Experiments where saddle coils are used to cancel the perturbation of the RWM have been carried out in the DIII-D tokamak. So far it has been possible to extend the phase with slow rotation and high b, but the present experiments are still ultimately terminated by excessive RWM growth [37]. The DIII-D RWM stabili-sation system has been stepwise upgraded to include optimised sensors and active coils inside the vacuum vessel which will permit a faster reaction to the RWM in the near future.

SUMMARY

It has been shown that the operational space of tokamaks is largely determined by the occurrence of large scale MHD modes. While the limits to current and density are usually avoided because practical limitations occur before the ultimate MHD limit is reached, the b-limit set by NTMs in conventional scenarii and RWMs in advanced scenarii are too low to guarantee economic operation of future fusion reactors based on the tokamak principle. Thus, means for active control of these two instabilities have been developed, based on an increased physics understanding of the MHD processes involved in these instabilities.

NTMs can be effectively removed by direct current drive within the island using ECCD. Complete suppression of both (3,2) and (2,1) NTMs has been demon-started in several tokamak experiments. Due to the sensitivity of the scheme to the correct deposition within the island, feedback control of the deposition is ultimately necessary. Several demonstrations of such a control exist. Further work will focus on the extrapolation of the requirements for NTM stabilisation in ITER.

RWMs can be avoided by sufficient plasma rotation and operation above the no-wall limit has been convincingly demonstrated. However, marginally stable RWMs tend to amplify error fields once the no-wall limit has been exceeded, which in turn slows down plasma rotation and lets the RWM penetrate the wall. Thus, ultimate control of the RWM may require active stabilisation by saddle coils. Experiments to demonstrate this technique are under way.

In summary, recent progress in active stabilisation of MHD instabilities has enabled tokamaks to be operated above the practical β-limits set by these insta-bilities. An increased understanding of the physics governing these processes has been obtained and predictions of how these techniques will work in future reactor-scale experiments are based on this. However, further work is clearly needed to create a firm physics base which allows safe extrapolation. Thus, this field will remain a primary focus of both theoretical and experimental work in tokamak MHD stability for the coming years.

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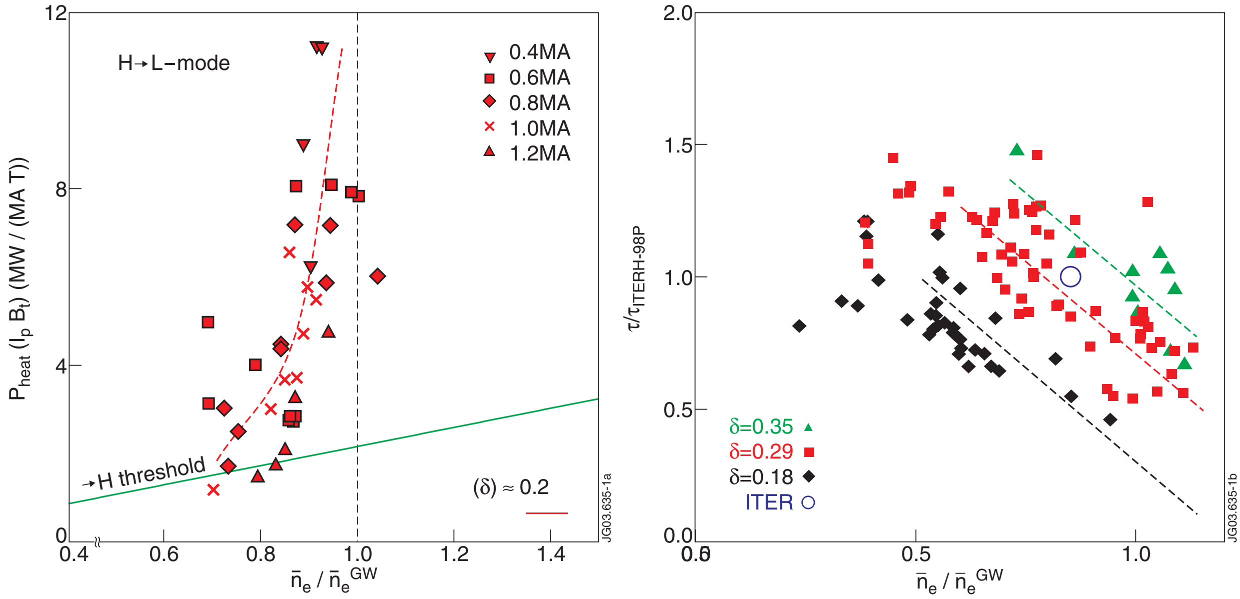


Figure 1: Practical density limit due to confinement degradation: the left hand side shows the power needed to stay in H-mode in ASDEX Upgrade compared to the usual threshold scaling derived for medium density. As n approaches n_{GW}, the required power increases excessively. On the right hand side, the confinement degradation of the H-mode itself when n approaches nGW is shown. Increasing triangularity increases the operational window for H-mode with good confinement.

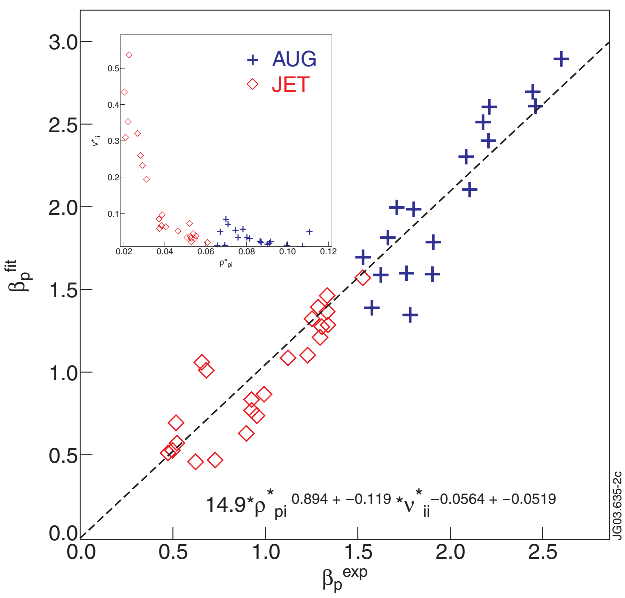


Figure 2: Scaling of the oset β_N for (3,2) NTMs in ASDEX Upgrade and JET. By proper normalisation of the quantities involved, an almost linear ρ^*_p -scaling with weal collisionality dependence is found for both machines.

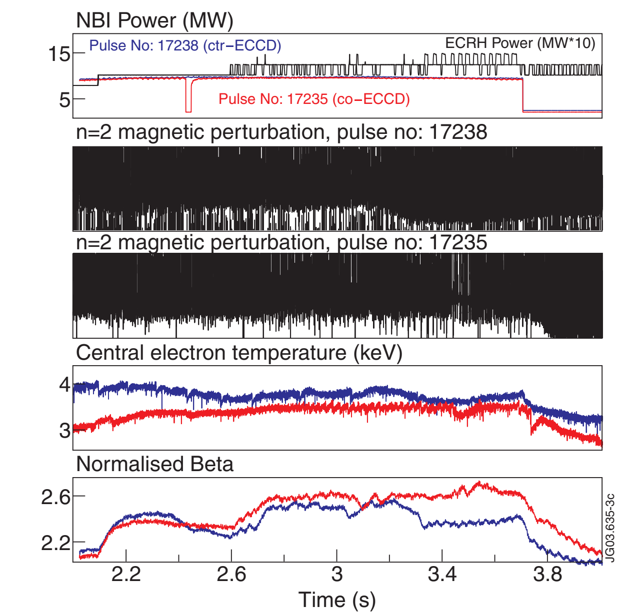


Figure 3: NTM avoidance by sawtooth tailoring using ECCD in ASDEX Upgrade: With rapid small sawteeth, no NTM appears whereas with big isolated sawteeth, an NTM is triggered.

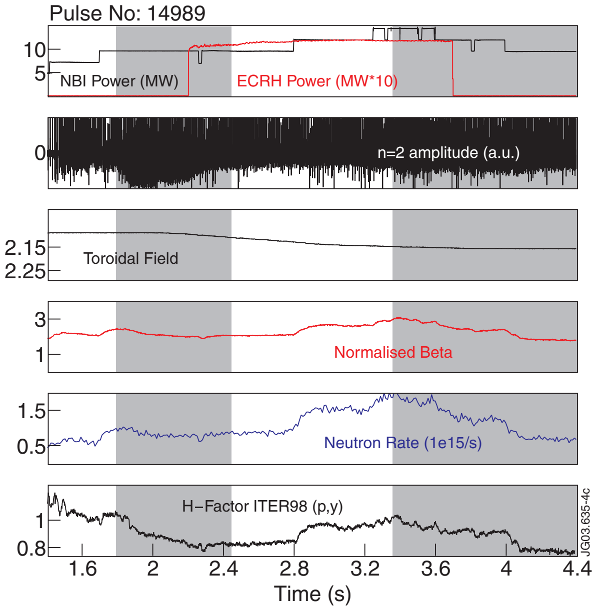


Figure 4: NTMstabilisation by DC ECCD on the high field side. The phases in which the (3,2) NTM exists are marked by the shaded areas. With increased NBI power, β_N can be raised significantly above the onset value and confinement recovers.

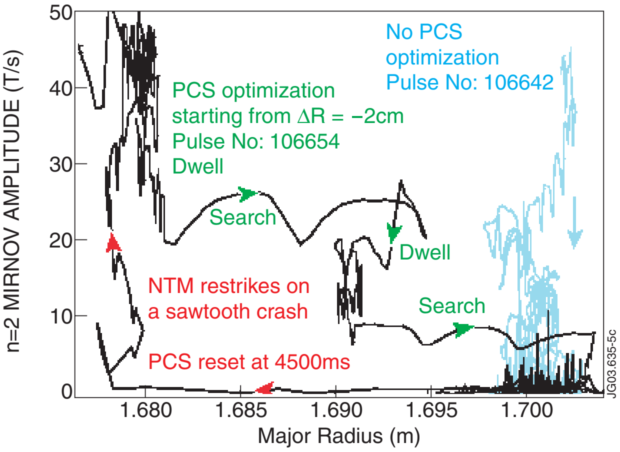
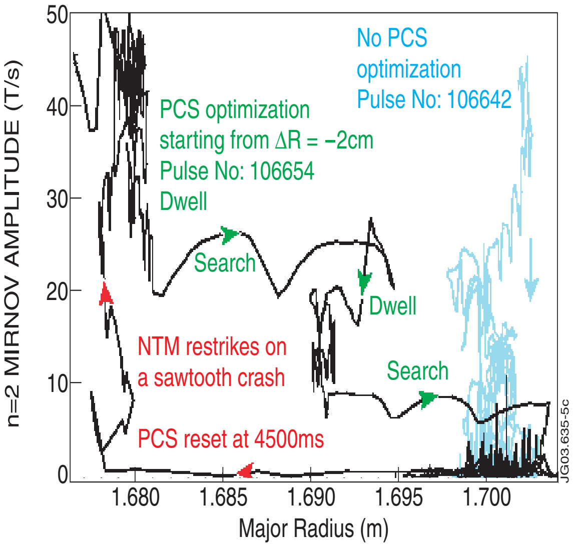


Figure 5: 'Search and suppress' algorithm on DIII-D to optimise ECCD deposition for NTM stabilisation. The major radius is adjusted in a stepwise manner to change the ECCD deposition. Depending on the effect on the mode amplitude, the next step is taken to further minimise the mode amplitude.



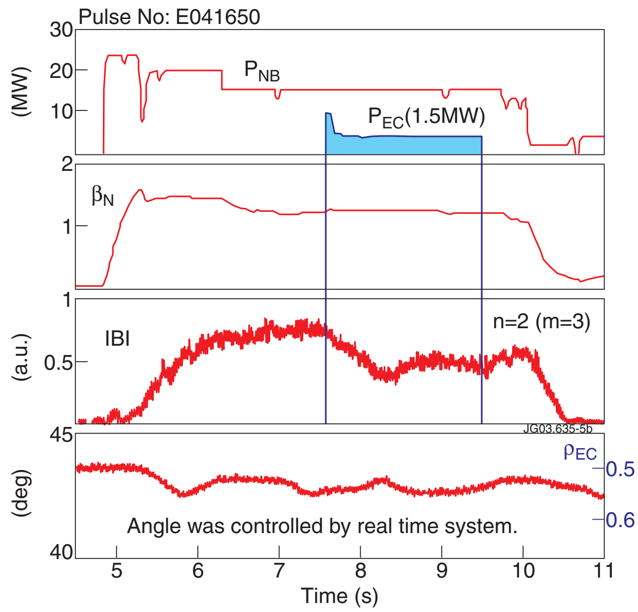


Figure 6: NTM stabilisation by DC ECCD in JT-60U. In the left figure, ECCD has been applied before NTM onset and the mode, once triggered, only grows to small amplitude. In the right figure, the same ECCD power cannot reduce the mode to an equally small amplitude when applied to a saturated mode.

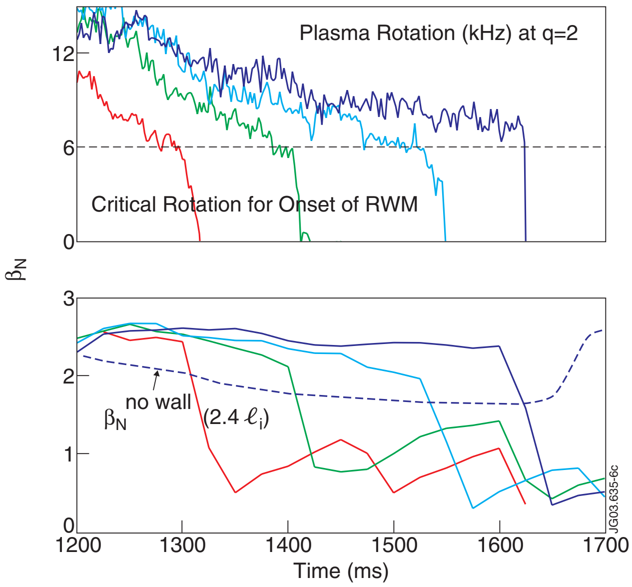


Figure 7: RWM stabilisation by rotation in DIII-D: As long as the plasma rotates above a certain threshold value, the no-wall β-limit can be significantly exceeded.