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ABSTRACT.
Key parts of the ITER scenarios are determined by the capability of the proposed poloidal field coil set. They include the plasma breakdown at low loop voltage, the current rise phase, the performance during the flat top phase, and a ramp down of the plasma. The ITER discharge evolution has been verified in dedicated experiments. New data are obtained from C-Mod, ASDEX Upgrade, DIII-D, JT-60U and JET. Results show that breakdown for E_{axis}<0.23-0.33V/m is possible un-assisted (ohmic) for large devices like JET and attainable in devices with a capability of using ECRH assist. For the current ramp up, good control of the plasma inductance is obtained using a full bore plasma shape with early X-point formation. This allows optimisation of the flux usage from the poloidal field set. Additional heating keeps $l_i(3)<0.85$ during the ramp up to $q_{95} = 3$. A rise phase with an H-mode transition is capable of achieving $l_i(3)<0.7$ at the start of the flat top. Operation of the H-mode reference scenario at $q_{95}\sim3$ and the hybrid scenario at q_{95} = 4-4.5 during the flat top phase is documented, providing data for the $l_i(3)$ evolution after the H-mode transition and the $l_i(3)$ evolution after a back-transition to L-mode. During the ITER ramp down it is important to remain diverted and to reduce the elongation. The inductance could be kept ≤1.2 during the first half of the current decay, using a slow I_p ramp-down, but still consuming flux from the transformer. Alternatively, the discharges can be kept in H-mode during most of the ramp down, requiring significant amounts of additional heating.

1. INTRODUCTION

Simulations and experiments are focused on 15MA scenarios for ITER [1], being the most challenging of the ITER reference scenarios for the superconducting Poloidal Field (PF) coils. In the course of producing suitable plasma configurations at 15MA in ITER, with sufficient wall clearance and control over the divertor strike point positions, the PF coils must remain within several limits, such as coil current, coil field, voltage, power, and central solenoid force limits. Recent studies [2, 3] have concentrated on upgrading the originally proposed PF coil set to provide better control and to respond to plasma disturbances within a range of plasma inductance (used here $l_i(3) = 2\int B_p^2 dV/(\mu_0^2 I_p^2 R_0)$, with B_p the poloidal magnetic field, I_p the plasma current, V plasma volume and R0 the major radius). Allowing for control margins [2, 4], a range of $l_i(3) = 0.7-1.0$ is possible for the current rise and flat top phase of ITER discharges at 15MA.

Until recently, detailed experimental data on the time evolution of ITER-like plasma discharges were not available. Moreover, the analyses performed in the framework of the ITER design review (2006−2008) highlighted that some of the assumptions made in the scenario simulations, in particular the evolution of the plasma inductance, are not consistent with experimental observations. Hence, dedicated experiments at C−Mod [5], ASDEX Upgrade [6], DIII-D [7] and JET [8] have been performed on all aspects of the discharge scenario. These dedicated experiments have (in part) been coordinated by the Steady State Operation Topic Group of the International Tokamak Physics Activity. They are also supported by interpretation of the plasma discharges with several scenario modelling codes [9,10].

The focus in this paper is on the control of li(3) during the discharge, to stay within the projected ITER limitations, and to provide a range of target q-profiles for the burn phase in standard ($q_{95} = 3$) and advanced scenarios (q_{95}>3). The results presented cover four phases of the ITER discharge scenario as given in Figure 1: (1) The plasma breakdown phase at low loop voltage, (2) the current rise phase of the discharge with variation of the plasma shape, plasma current ramp rate, and heating used, (3) the current flat top phase of the H-mode reference scenario at q_{95}~3 as well as the hybrid scenario at $q_{95} = 4-4.5$ and finally (4) the current ramp down phase which has to terminate the discharge safely whilst maintaining a vertically stable plasma equilibrium. Details of the results obtained in JET and ASDEX Upgrade are given here. The results of JT-60U, C-Mod and DIII-D are summarised.

2. LOW VOLTAGE BREAKDOWN EXPERIMENTS

Most devices have revisited low voltage plasma breakdown recently to match ITER conditions, i.e. having available an electrical field of 0.33V/m on axis (E_{axis}). Both JET and DIII-D [11] have optimised low voltage start-up. ASDEX Upgrade and Tore Supra [12] developed, for the first time, operation without resistor switches in the ohmic heating circuits. Several superconducting tokamaks contributed to these studies (Tore Supra, EAST [13] and KSTAR [14]). Table I lists devices that performed these experiments, together with their size (major radius), the toroidal field used, and the type and amount of heating assist available. An overview is given on the values of electric field used to get reliable breakdown in these experiments for ohmic conditions and assisted plasma breakdown using Electron Cyclotron Heating (ECRH) or Lower Hybrid Current Drive (LHCD). Note that further optimisation of the breakdown phase may give even lower values for the electric field required for reliable plasma breakdown [see for example [15]). The results show that for un-assisted (ohmic) breakdown, the minimum achieved electric field on-axis tends to decrease with machine size down to ~0.23V/m for JET, a value well below ITER design value (0.33V/m). The result listed from JT-60U (1997), are in conditions not optimised for achieving the lowest voltage for ohmic breakdown. Most experiments have also tested ECRH breakdown assist, observing pre-ionisation of the filling gas. JET uses LHCD but observes no pre-ionisation. Pre-ionisation and applying additional heating during the plasma current rise to 100kA−200kA (“burn-through” phase), allows a reduction of the loop voltage required for reliable breakdown, with all devices achieving plasma breakdown at Eaxis~0.2-0.3V/m in clean machine conditions.

JET and ASDEX Upgrade performed several studies for breakdown at low loop voltage and are described in detail below. Standard low voltage breakdown settings for JET use zero shaping, giving an open field line configuration with stray field B_{stray}<0.4mT anywhere inside the vacuum chamber; the value on-axis being ~0.3mT. Shaping is added to form a well defined hexapole field null configuration with a stray field on-axis of $B_{stray} = 0.1$mT (see Figure 2a). The level of shaping can be varied, controlling the radial extent of the low stray field region (“size of the field null”) and the connection length. For the field null configuration shown in Figure 2a, the initial position of the plasma is on the high field side ($R<R_0$, JET = 2.96m). In these conditions the stray field near the inner wall is still low enough (B_{stray}~0.7-0.8mT) to have plasma breakdown at the higher electric field available at smaller major radius. For smaller field null configurations (higher shaping) the breakdown is delayed, due to reduced connection length and the plasma starts more outwards by 20-40cm. In these conditions however, the initial position of the plasma can be shifted more inboard by increasing the vertical field during plasma breakdown. Results show reliable breakdown without assist (ohmic) down to 0.23V/m. Applying 1MW LHCD, no pre-ionisation is observed of the filling gas is observed. LHCD assists the plasma current rise phase to 100kA–200kA (earlier mentioned "burn-through" phase), achieving reliable breakdown down to 0.19V/m. In reducing the available loop voltage, the plasma breakdown is delayed by 50ms–100ms (Figure 3a) and a slow and linear rise of the plasma current is observed. Using low voltage schemes, breakdown is still achievable after a high current (q_{95} = 3) disruption (previous pulse). By adjusting the prefill at 0.33V/m and using 1MW LHCD assist, plasma breakdown is successful, albeit at somewhat higher plasma density with a resulting slower rise of the plasma current compared to clean vessel conditions.

Also ASDEX Upgrade uses a hexapole field null configuration (Figure 2b) for low voltage breakdown studies (E_{axis}~0.2-0.5V/m). In all cases ECRH is applied to pre-ionise the filling gas. The ECRH resonance position is near $R = R_0$, AUG = 1.65m, using second harmonic X-mode heating at 140GHz, for 2.5T and 105GHz at 1.7T. Pre-ionisation is observed for P_{ECRH} > 300kW, with more prompt ionisation at higher input power, the pre-ionisation at 105GHz being more efficient (earlier at the same input power) compared to 140GHz. Similar to the results obtained in JET, a scan of the loop voltage from the OH circuit show that reliable breakdown can be obtained down to E_{axis} ~ 0.21V/m, with a marked slow down of the initial current rise at lower loop voltage (Figure 3b). Using 105GHz at 1.7T, the resonance position is scanned from a minimum of 1.40m to a maximum of 1.85m (R_0, AUG = 1.65m). At the extreme positions of the scan, the resonance lies outside the field null (minimum B_{stray}), and the plasma breakdown can not be sustained. A demonstration using fundamental O-mode heating of 105GHz at 3.1T shows an increase in plasma density directly after application of the ECRH power during the pre-ionisation phase. Also the current rise is faster compared to second harmonic X-mode heating at 1.7T. At 3.1T, the ECRH resonance position is at 1.45m. Despite this mismatch between the field null position and the ECRH resonance position, the plasma breakdown is reliable using fundamental O-mode injection. With the resonance position of ECRH on the high field side, these discharges at ASDEX Upgrade using 105GHz at 3.1T are identical to using 170GHz at 5.2-5.3T in ITER. Hence, a dedicated ECRH system at 127GHz would not be essential for ITER.

JT-60U [18] has not optimised specifically for ITER-like breakdown conditions in recent experiments but uses routinely 2MW ECRH to achieve robust breakdown, even in successive high recycling discharges.

All experiments observe a decrease of the initial (first 100–200ms) rate of rise of the plasma current going to lower loop voltage. Typical values for this initial rate of current rise vary from 0.5 to 1.3MA/s at Eaxis~0.2V/m. The slow rise allows current penetration without MHD reconnection, giving access to low li(3)~0.3-0.6 just after breakdown. Hence, low voltage breakdown settings are used in most of the ITER scenario demonstration discharges (described below).

3. CURRENT RISE PHASE

One of the main aims of dedicated experiments is to demonstrate operation with $0.7<l_i(3)<1.0$ throughout the current rise phase, ramping to $q_{95} \sim 3$ (high normalised current). The experiments are designed (scaled down from ITER, Figure 1) using the plasma resistivity as guide (resistivity~$<T_e>^{3/2} a_{min}^2$, where $<T_e>$ is the volume averaged electron temperature and amin the minor radius). Current rise studies at $q_{95} = 3$ in ASDEX Upgrade operate at 1.0MA/1.7T reaching full current in 1.0s-1.2s. JET uses 2.7MA/2.4T, with a rise phase variable from 6s to 10s. DIII-D ramps to 1.64MA at 2.14 T within 1.6s and C-Mod uses a current ramp-up scheme to 1.35MA within 0.4-0.6s at 5.4T. Typically (except C-Mod) low voltage (E_{axis} = 0.2-0.3V/m) is used during the plasma breakdown phase of these dedicated experiments. The studies concentrate on four topics detailed below: (1) the optimum plasma shape evolution, (2) ohmic discharges, (3) use of additional heating, and (4) tools available for $l_i(3)$ control.

PLASMA SHAPE:

The original startup scenario envisioned for ITER [19] starts with a small outboard limited plasma. The plasma cross section is expanded to keep constant q at the plasma boundary as the plasma current increases, diverting at 7.5MA. Experiments duplicating this scenario shows a rapid (as designed) current penetration during the limiter phase, featuring high $l_i(3)>1$, just before X-point formation. DIII-D, ASDEX Upgrade, C-Mod and JET demonstrate that low plasma inductance is only achieved with a full bore limiter phase (limited on the outboard side to reproduce ITER conditions) and diverting as early as possible. This also allows early use of additional heating, during the divertor phase. Figure 4 shows results obtained in JET and DIII-D [11], comparing different cross section size during the early ohmic ramp-up phase. Three ohmic discharges at JET are compared with different plasma size during the limiter phase, and different timing of the X-point formation (Figure 4a). The JET discharges (e.g. Pulse No: 72467) with maximum aperture during the limiter phase and early X-point formation readily achieve low li(3) during the first part of the current rise phase, having hollow temperature profiles. All experiments show excellent reproducibility of the full bore limiter, early X-point scenario, with good control of the plasma density.

OHMIC DISCHARGES:

In JET, the density is varied in ohmic conditions from very low density ($<n_e>/nG_W \sim 0.2$, with $<n_e>$ the line averaged electron density and n_{GW} the Greenwald density limit) to intermediate density ($<n_e>/nG_W \sim 0.4$). These discharges show no variation of $l_i(3)$ as the effective charge of the plasma, Z_{eff}, reduces at higher density from 1.6 to 1.2 respectively, while $<T_e>$ decreases by 25%, leading to similar plasma resistivity. Increasing the density further during the current rise to $<n_e>/n_{GW} \sim 0.6$ (as done in experiments in ASDEX Upgrade), maintains Z_{eff} at 1.2–1.3 while $<T_e>$ decreases with $<n_e>$, leading to higher values for $l_i(3)$. Hence, in ohmic conditions experiments observe a clear optimum for the current diffusion; a trade-off between achieving high Te at low density ($<n_e>/n_{GW} < 0.2$) but rather higher Z_{eff} ~1.5-2.5, or somewhat higher density ($<n_e>/n_{GW}$~0.4) at reduced T_e but significantly lower Z_{eff} ~1.2-1.5. In general, the results show that stable ohmic discharges at $q_{95} \sim 3$ have the lowest $l_i(3) = 0.8–0.85$ when using the fastest current ramp rates available after the breakdown phase. As shown in Figure 5a, the Ip ramp rate is varied in JET from 0.36MA/s to 0.19MA/s, giving a variation of $l_i(3) = 0.83$-1.03, ASDEX Upgrade varies dIp/dt from 0.92MA/s to 0.66MA/s giving $l_i(3) = 0.82$-1.0. C-Mod changes dIp/dt from 2.4MA/s to 1.3MA/s giving $l_i(3) = 0.9$-1.0. These results extrapolate to ITER having a fast current rise to 15MA of ~70s and a slow rise of ~100s. DIII-D can obtain l_i~0.65, but these discharges are MHD unstable leading to disruptions at full current ($q_{95} \sim 3$). Moreover, they extrapolate to a ramp up of 50s in ITER, too fast for the proposed PF power supplies. During the flat top without additional heating, $l_i(3)$ increases to 1.1-1.2. In ITER such high $l_i(3)>1$ is not accessible at 15MA with the available flux from the OH transformer [2,4], implying that ITER will have to start heating, at the latest, immediately after reaching 15MA.

ADDITIONAL HEATING:

Heating during the limiter phase can use the inboard or outboard limiters of the device for power handling. All experiments observe a rapid increase of Z_{eff} when using additional heating. For example in ASDEX Upgrade (W-wall, using the outboard limiters) Z_{eff} raises to 2-3, similar to observations in C-Mod (Mo-wall, touching both inboard and outboard limiters), while Zeff reaches ~4 in JET (C-wall, Be coated, outboard limiters). In the various experiments, the type and level of heating during the divertor phase of the current rise is varied. ASDEX Upgrade uses Neutral Beam Injection (NBI) with on-axis and off axis sources (1.5–5MW) or ECRH at 0.5 MW. JET applies both on axis or off-axis ion- cyclotron resonance heating, ICRH, at 2 to 6MW, or LHCD up to 2.2MW or NBI up to 10MW. DIII-D utilises NBI (1–5MW) and C-Mod uses central ICRH (1–3MW). A clear result is that heating during the current rise, in L-mode or in H-mode, gives a capability of significantly varying $l_i(3)$ from 0.97 to 0.63 at fixed dI_p/dt. An overview of all experiments is given in Figure 5b. Details for ASDEX and JET are shown in Figures 6a and 6b respectively. In L-mode, $l_i(3)$ values as low as 0.8 are reached. JET shows no difference in the $l_i(3)$ achieved at $q_{95} = 3$, using 3MW central ICRH, or 2.2MW LHCD or 4MW NBI. Within the range of heating power available, transitions to H-mode are observed in DIII-D, ASDEX Upgrade and JET, giving access the lowest $l_i(3)$=0.63-0.75 with reversed q-profiles. Code simulations [2] show that the heating effect on $l_i(3)$ dominates over any current drive effect from either NBI or

LHCD. TRANSP/TCS simulations of the current rise phase in C-Mod show that comparing an ohmic current rise with a 2MW ICRF heated case, the flux saving is due to a reduction of ~20% in the resistive flux consumption as shown in Figure 7. Likewise in ASDEX Upgrade and JET, discharges with an H-mode current rise phase save 25%-30% of the transformer flux required for an ohmic current rise. In H-mode, the bootstrap current near the pedestal plays an important role. In addition, a broad Te profile helps in forming broad current density profiles. In ASDEX Upgrade for example, target plasmas with $l_i \sim 0.63$ at $q_{95} = 3$ are used for the hybrid regime exploration at low q_{95}.

CONTROL OF LI(3):

At DIII-D, feedback control of $l_i(3)$ is employed during the divertor phase of the current rise of large-bore startup discharges, using the current ramp rate as the means of changing $l_i(3)$ [10]. The ramp rate is varied from 0.34MA/s to 1.5MA/s. Control of li(3) in purely inductive (ohmic) current rise experiments and with various levels of NBI during the current rise is demonstrated. As expected, the inductive cases without heating require higher current rampup rates to achieve lower $l_i(3)$. Increasing levels of auxiliary heating lead to slower current ramp up rates to maintain the same level of l_i. More sophisticated control schemes using density, heating, and current ramp rate are under development in DIII-D [20] for generating a specified q profile. At JET, control of $l_i(3)$ by additional heating is applied in scenarios with a current rise to $q_{95} = 4$ (2MA/2.4T). Control is demonstrated with either ICRH or NBI. Requesting $l_i(3) = 0.8$, a target q-profile with q(0) just above 1 at the start of the flat top is produced requiring modest heating powers (ICRH~3MW, NBI~5MW), these results are shown in Figure 8. DIII-D and JET have shown that at even lower I_p with q_{95} near 5, central q values near 2 can be produced in an ITER like current rise. This is required as a target for advanced scenarios with the aim of producing Q~5 in full steady state conditions.

4. PERFORMANCE DURING THE FLAT TOP PHASE

In ITER, the "nominal" 15 MA ELMy H-mode plasma is characterized by $I_p = 15$MA, $B_T = 5.3$T, $R_0 = 6.2$m, $a_{min} = 2.0$m, elongation; $\kappa = 1.85$, $\langle n_e\rangle/n_{GW} = 0.85$, $l_i(3) = 0.8$, poloidal beta; $\beta_p = 0.8$, normalised beta; $\beta_N = 1.8$, alpha heating power; P±=80MW, additional heating; $P_{aux} = 40$MW, and PL-H = 80MW (H-mode power threshold in a 50.50 DT mix, using the latest threshold scaling law [21]). The majority of the devices studying ITER relevant ramp-up scenarios continue the studies during the flat top phase for the H-mode inductive scenario at $q_{95} \sim 3$. The experiments aim at obtaining an enhancement factor over the ITER98y2 scaling law [19], H98~1 and $\beta_N \sim 1.8$. Apart from C-Mod (ICRH), the dominant heating power in these experiments is neutral beam heating, although typically $T_i(0) \sim T_e(0)$ is obtained in these discharges. In C-Mod, DIII-D and JET, $\langle n_e\rangle/n_{GW} = 0.6$-$0.65$ is achieved during the flat top phase without additional gas fuelling. ASDEX Upgrade obtains $\langle n_e\rangle/n_{GW} = 0.78$ using deuterium gas fuelling ($\Phi_D = 8 \cdot 10^{20}$/s). No active ELM mitigation or radiation seeding is used in these discharges. Table II gives an overview of the results.

A few specific issues are documented during the flat top phase: (1) The evolution of the plasma inductance, (2) entry into a stationary H-mode phase and (3) the discharge evolution following a back-transition to L-mode.

After the transition to H-mode, the experiments extend the heating phase to several resistive diffusion times (τ_R) during flat top (limited by the magnet coils and/or pulse length of the additional heating systems). The maximum pulse length in these experiments is ~2-3τ_R for ASDEX Upgrade, ~3τ_R for DIII-D and ~1-1.5τ_R for JET. The l_i(3) value at the end of the Flat Top (FT), is given in the last column of Table II. Figure 9a shows the l_i(3) evolution for the current rise and flat top phase of the discharges given in Table II. During H-mode most experiments observe a slow evolution of l_i(3) to values ≤0.85. The value at the end of the flat top phase is independent of the starting values at the beginning of the current flat top. The discharges for DIII-D and JET shown in Figure 9a have a current rise giving $l_i(3) = 0.85$-0.9. If required, the current rise can be controlled (heating power) to give the same l_i(3) value at the start of the flat top as the end of the flat top. DIII-D and C-Mod both matched the ITER shape, having low ELM frequency or long ELM free periods, DIII-D achieving the lowest values for $l_i \sim 0.65$ in these conditions, as reported in reference [22].

Energy confinement factors of $H_{98} \sim 1$ are obtained as necessary for ITER. The input power level required to obtain $\beta_N \sim 1.8$ is compared to the latest H-mode scaling [21], showing a wide range in the experiments from to $(1.0-2.1) \cdot P_{L-H}$, the highest values observed in JET. The values predicted for ITER lie within this range, having a total heating power (including α-power) of $(1.1$-$1.5) \cdot P_{L-H}$. After entering H-mode, the experiments take ~2 energy confinement times (τ_R) to reach maximum stored energy and a minimum of ~4-6τ_E to reach stationary electron density values. Figure 9b shows DIII-D and JET discharges that have a deliberate power step down to provoke a back-transition to L-mode during the flat top phase. For DIII-D the neutral beams are turned off at 3.5s, inducing a H-L back-transition at 3.73s (after an ELM-free phase), followed by a disruption at 3.86s. The JET discharge reduces NBI from 17MW to 3MW at 10s, showing that l_i(3) rises to ~1.0 within 3s.

HYBRID SCENARIO:

Experiments have extended studies of ITER scenario demonstrations (breakdown, rise phase and flat top) to $q_{95} = 4$-4.5. All, including C-Mod which uses LHCD [23], show that the required target q-profile with q(0) just above or near 1 is obtained. Low magnetic shear has been achieved in the core in DIII-D and ASDEX Upgrade. High beta and high confinement properties are observed in ASDEX Upgrade, DIII-D, JET and JT-60U. In these experiments, hybrid discharges obtaining $\beta_N \sim 3$ have $l_i(3) = 0.6-0.75$. In the demonstration discharges, both DIII-D and ASDEX Upgrade have $1.2 < H_{98} < 1.45$ capable of achieving Q~10 in ITER at $q_{95} = 4$-4.5. The confinement is documented for a range of conditions including the lowest ρ^* values obtained in JET and JT-60U. New JET results show that $H_{98} = 1.2$-1.4 can be obtained [24]. Long pulse capability is demonstrated in JT-60U, sustaining $\beta_N \sim 2.6$ and $H_{98} > 1$ for 25 seconds at somewhat lower

q_{95}~3.2 [18]. More data are required from ITER hybrid scenario studies, focussing on achieving H_{98}>1.2 at T_i/T_e and low plasma rotation. Dependencies on these parameters are highlighted by DIII-D experiments [25]. Planned experiments in C-Mod and JET for 2009 are important in this regard.

5. CURRENT DECAY PHASE

Although detailed evolutions (model calculations) of the discharge ramp-down phase of ITER have not yet been completed, initial experimental work has also studied discharge shut-down scenarios. As shown in an example from ASDEX (Figure 10), the current ramp down phase must provide a (vertically) stable ramp down of the plasma current, transitioning from H-mode to L-mode, allowing control over the radiation fraction, keeping below the density limit and avoiding overheating of first wall components. In ITER additonal contraints are staying within the available full swing of the transformer and exiting the burn phase.

C-Mod, ASDEX Upgrade, DIII-D and JET have developed ramp down scenarios that keep the plasma diverted as long as possible, using an elongation reduction (from 1.85 to 1.5) to keep the plasma vertically stable. So far the experiments concentrate on documenting the requirements for keeping $l_i(3)<1.6$ before 50% of the flat top current value is reached. At $l_i(3)>1.6$ and high plasma current, the growth rates for vertical displacements probably can not be stabilized in ITER, although more detailed studies are needed. All experiments show that for ohmic or L-mode plasmas $l_i(3)$ rises above 1.6 for moderate to fast ramp down rates. Examples from C-Mod are given in Figure 11. Only a 1MA/s ramp down (slow for C-Mod) keeps $l_i(3)$ below 1.6 (note this discharge did not have a reduction in elongation to keep the plasma vertically stable). However, this slow ramp down requires an additional 10% of transformer current as indicated in Figure 11c. At JET, ohmic ramp down discharges at 0.28MA/s, keeping constant current in the transformer, show an increase of $l_i(3)$ to 1.8. Consequently, scenarios that maintain H-mode throughout the ramp down phase have been studied. Preliminary results from ASDEX Upgrade, DIII-D and JET show that the current can be ramped down without additional flux consumption while keeping $l_i(3)$ low enough. However, H-mode can only be kept throughout the current decay phase with constant heating at a level of >50% of the heating required during the flat top phase and at relatively slow current ramp down rates. The requirements for the ramp down seem challenging for ITER; hence a modelling effort for the decay phase of ITER using these new experimental data is urgently required. Furthermore, significant levels of additional heating may be required until the current has reached I_p ~ 3MA in ITER.

SUMMARY AND CONCLUSIONS

The experimental verification of ITER scenarios has provided new data for all phases of the discharge. They include studies of the plasma breakdown at low voltage. These show that the minimum electric field for reliable ohmic (un-assisted) breakdown decreases with machine size to values of ~0.23V/m in JET. For assisted breakdown, using ITER relevant ECRH schemes, all experiments using this technique have established reliable breakdown at or below ITER values of 0.33V/m in clean or de-conditioned machine circumstances. The current rise phase has been studied in detail in these new experiments. Ramping to $q_{95} = 3$, the current profile can be tailored to obtain a large variation of the plasma inductance. It is strongly recommended to use full bore plasmas with early X-point formation during the current rise phase. Full bore plasma configurations give access to lower plasma inductance compared to small bore plasma start-up scenarios. However in a full bore early X-point formation scheme, high values ($l_i(3) = 1.05$) can still be accessed in ohmic discharges with a relatively slow current ramp up rate. The lowest $l_i(3) = 0.63$-0.68 is achieved in discharges heated to H-mode during the rise phase. Code simulations for ITER indicate [9] a requirement to divert at the time $I_p = 3.5$MA is reached and to subsequently heat during the current rise with $P_{aux} = 5$-15MW, depending on the li(3) values required. During the flat top phase experiments have reproduced the requirements for reaching $Q = 10$ at $q_{95} = 3$. Data on the evolution of the plasma parameters, in particular the slow evolution of the plasma inductance to values of 0.65-0.85, provide useful data for studying the requirements for the poloidal field coil set in ITER. The current decay phase deserves more evaluation. Experiments clearly show that in ohmic and L-mode conditions only a very slow current ramp down can keep $l_i(3)<1.6$ during the first half of the current decay. Translated to ITER a 300s ramp down phase would be required, likely to consume transformer flux (in such conditions, C-Mod requires 10% additional current in the main OH coil). Results from ramp down experiments in H-mode have been obtained recently, indicating the possibility to keep $l_i(3)$ low enough. However, the requirements for the heating systems to provide sufficient heating to stay in H-mode during most of the ramp down phase need to be assessed. Several areas for ITER scenario demonstration remain to be explored, such as burn control and RF-dominated heating schemes with low rotation. Advanced ITER scenarios will be the focus of future experiments.

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	R_0 [m]	B_T [T]	ECRH	Power (type)[(1)]	E (V/m) Ohmic	E (V/m) assisted
C-Mod	0.68	5.4	-	-	1.2-1.6	-
AUG	1.65	1.7-3.2	105-140 GHz	0.3-1 MW (X2,O1)	0.6	0.2
EAST	1.70	2.0	- (LHCD)	0.3 MW (LH)	0.5	0.3
DIII-D	1.70	1.9-2.1	110 GHz	1-1.4 MW (X2)	0.43[(2)]	0.21[(2)]
KSTAR	1.80	1.5	84 GHz	0.35 MW (X2)	-	0.4
TS	2.40	3.85	118 GHz	0.3-06 MW (O1)	0.3	0.15
JET	2.96	2.36	- (LHCD)	1.0-20 MW (LH)	0.23	0.18
JT-60U	3.32	3.5	110 GHz	0.4-2 MW (O1)	0.43[16]	0.26[17]
ITER	6.20	5.3	127 (170) GHz	3 (20) MW (O1)	≤0.33	≤0.33

(1): O1: Fundamental O-mode, X2: Second harmonic X-mode.
(2) In 1991, ref [15] documented values of 0.25V/m for ohmic and 0.15V/m for ECRH assist.

Table I: Recent ITER like low voltage breakdown studies.

	I_p [MA] /B_T[T]	P_{tot} [MW]	<n_e> [10^{19} m^{-3}]	β_p / β_N	H_{98}	f_{GW}	P_{tot}/P_{L-H} (1)	$l_i(3)$ (end of FT)
AUG	1.0 / 1.7	5.0	9.8	0.85 /1.9	0.95	0.78	1.5-1.7	0.85
DIII-D	1.5 / 1.9	4.5	8.0	0.65/1.8	1.0	0.65	1.0-1.5	0.65
JET	2.5 / 2.35	19.0	6.4	0.7 / 18	0.95-0.98	0.70	1.9-2.1	0.80
ITER	15 / 5.3	40+80 (2)	10.0	0.8 / 1.8	1.0	0.85	1.1-1.5 (3)	?

(1) P_{L-H} [MW] = $2.15 * n_{e20}^{0.782} * B_T^{0.772} * a^{0.975} R^{1.0}$ [ref 21, eq. (3)], the range indicated for P_{tot}/P_{L-H} is due to a rise in density during the H-mode phase.

(2) Projected α-power in ITER.

(3) For a 50:50 D-T mix.

Table II: Overview of ITER demonstration discharge parameters.

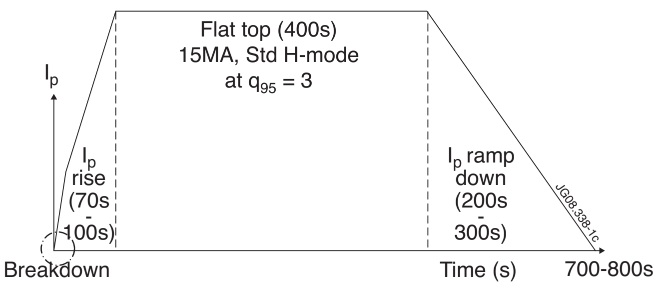


Figure 1: Sketch of an ITER discharge scenario with four phases of the discharge, (1) breakdown phase, (2) current rise phase (70-100s), (3) flat top phase (400s at 15MA and q_{95}=3) and (4) a slow (~200s-300s) current ramp down phase.

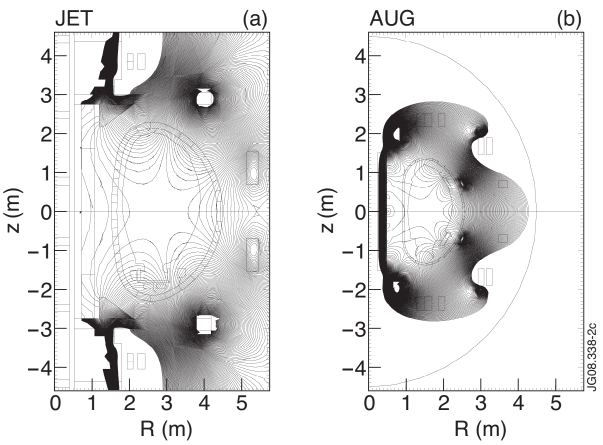


Figure 2: The contours of constant flux at breakdown for JET (a) and ASDEX Upgrade (b) plotted in combination with the cross-sections of the vacuum vessel and poloidal field coils. Both devices are shown on the same scale to demonstrate the difference in machine and field null size. The poloidal field coils are configured for both devices to provide a minimum stray field near the centre of the vacuum vessel.

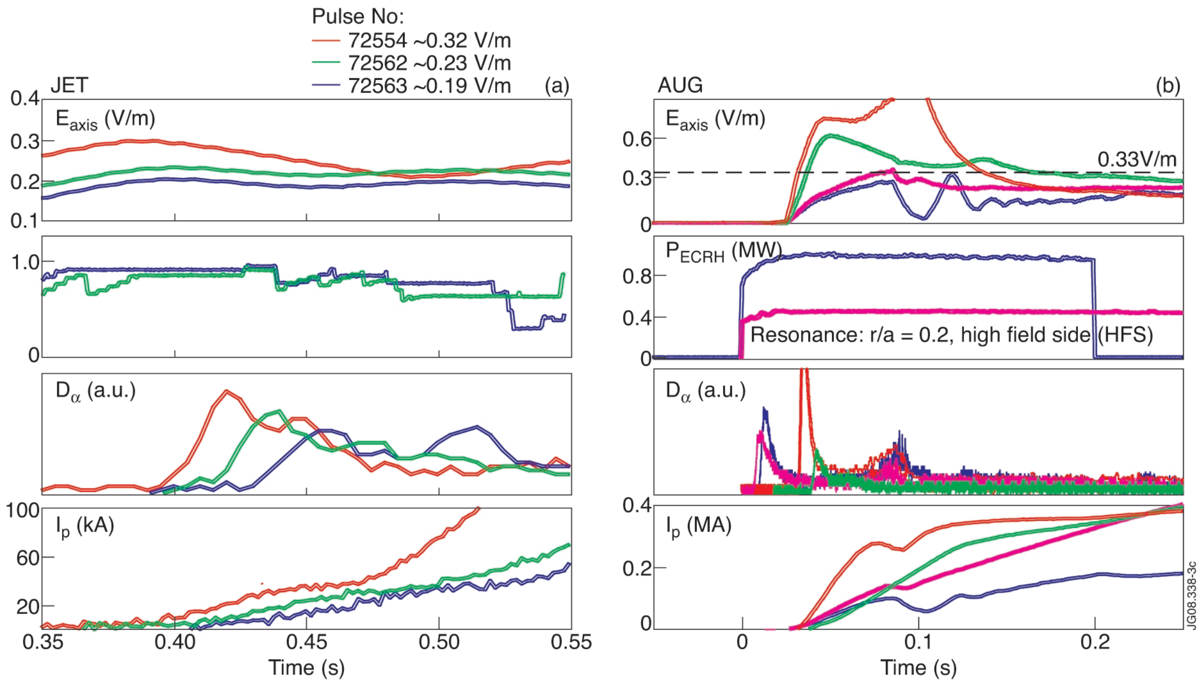


Figure 3: (a) Low voltage start up experiments in JET and (b) ECRH start up assist at ASDEX Upgrade. Shown from top to bottom are the electric field on axis, the LHCD or ECRH power used, the measurements of the $D\alpha$ emission in the main chamber, and the plasma current evolution. For JET, no pre-ionisation is observed (no D-alpha emission) before the plasma current rises. For ASDEX Upgrade, the minimum required electrical field on axis is reduced to ~0.2V/m using 0.3-0.9 MW ECRH (X2).

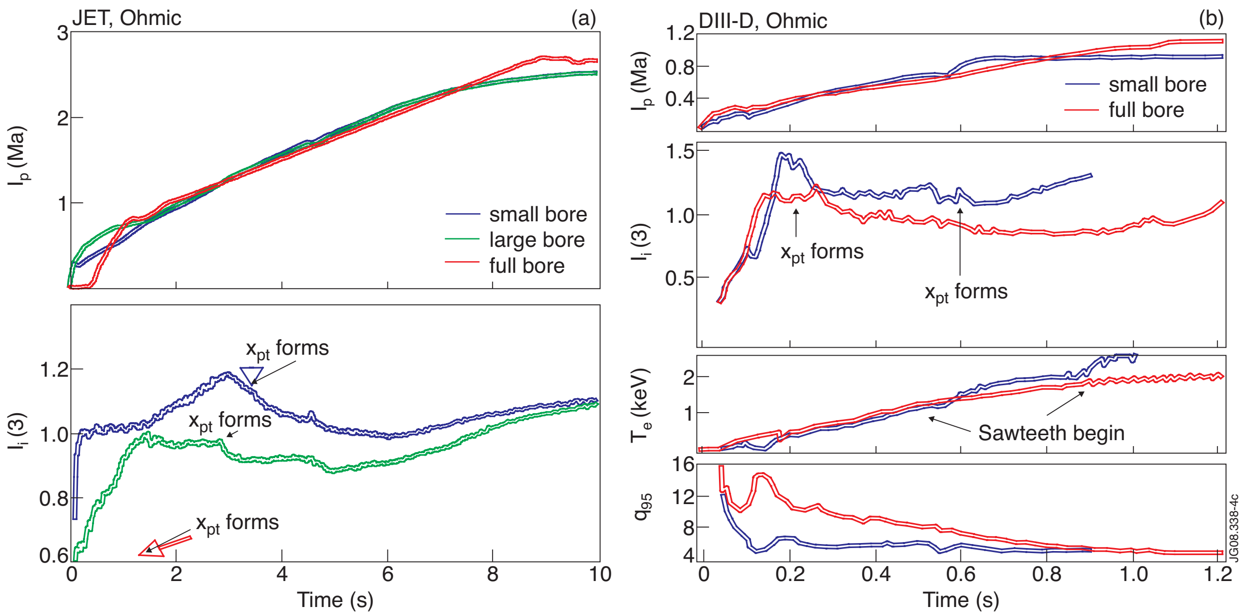


Figure 4: Evolution of $l_i(3)$ for ohmic current rise phases at JET (a) and DIII-D (b). The evolution for the originally envisaged small bore start up for ITER is indicated in blue. Full bore ramp up discharges for both devices are indicated in red. The green curve for JET is a large bore outer limiter case with somewhat later X-point formation compared to the red discharge.

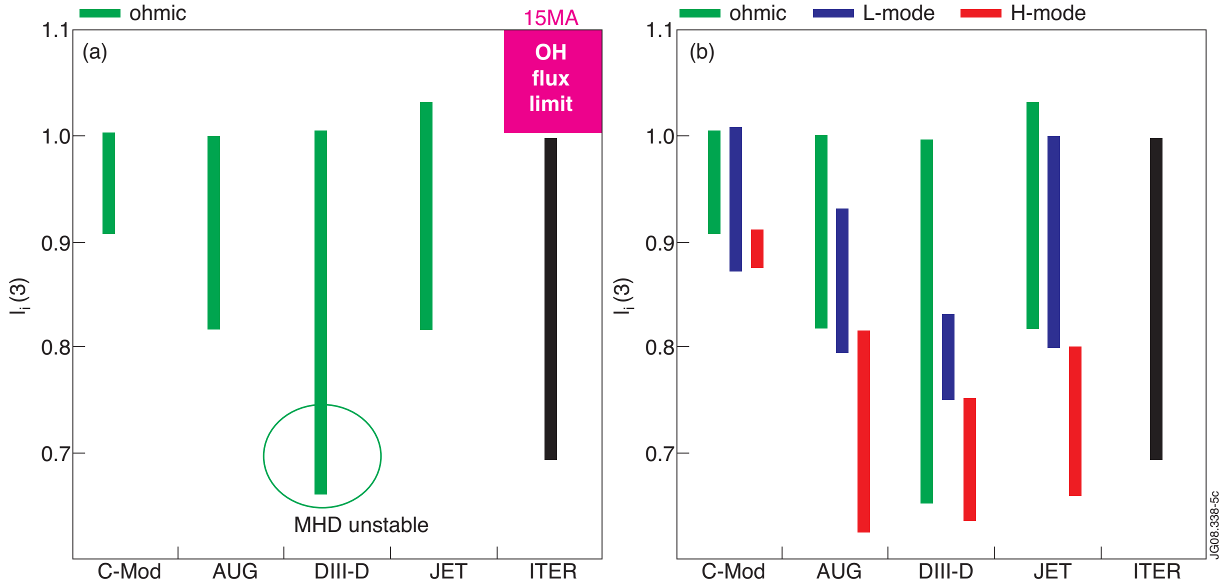


Figure 5*: Range of $l_i(3)$ obtained at the end of the current rise phase for various devices, for (a) ohmic current rise experiments and (b) compared to results obtained in heated discharges. In both graphs the ITER range for $l_i(3)$ is indicated by the black bars. Indicated by green bars are the values achieved during an ohmic current rise in various devices. Figure (a), gives an indication for the OH flux limit of ohmic current rise phases in ITER when $l_i(3)$ >1 as calculated in reference [2]. Added in figure (b) are results from L-mode (blue) and H-mode (red) discharges. Note that only a limited experimental data set is available for L-mode discharges in ASDEX Upgrade and DIII-D.*

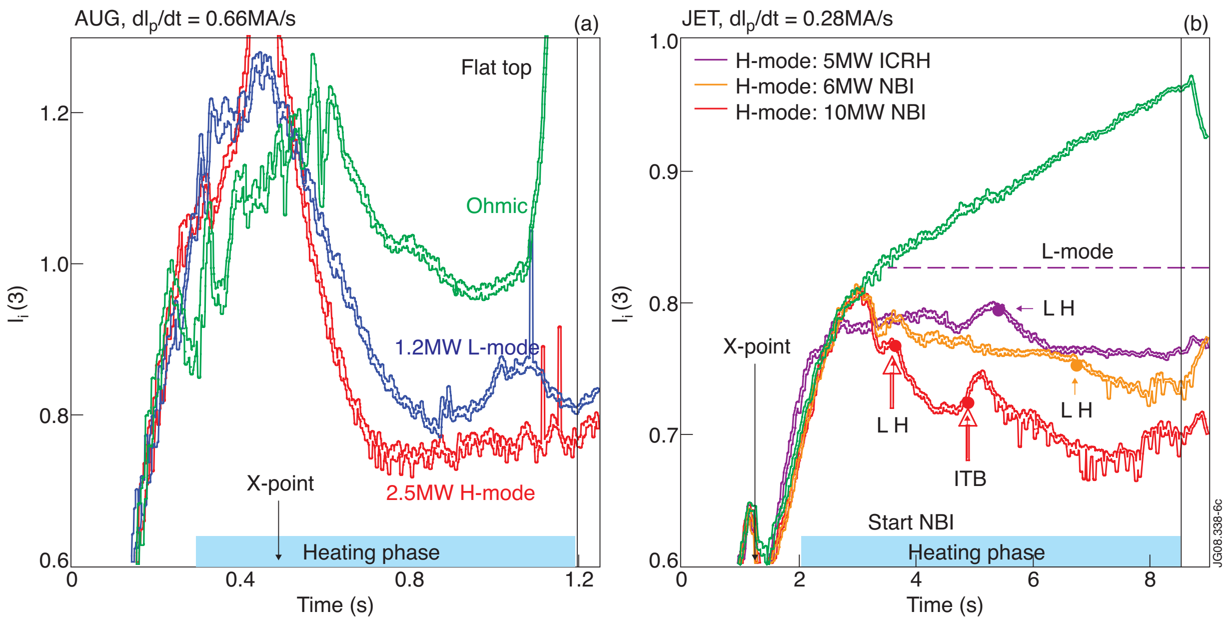
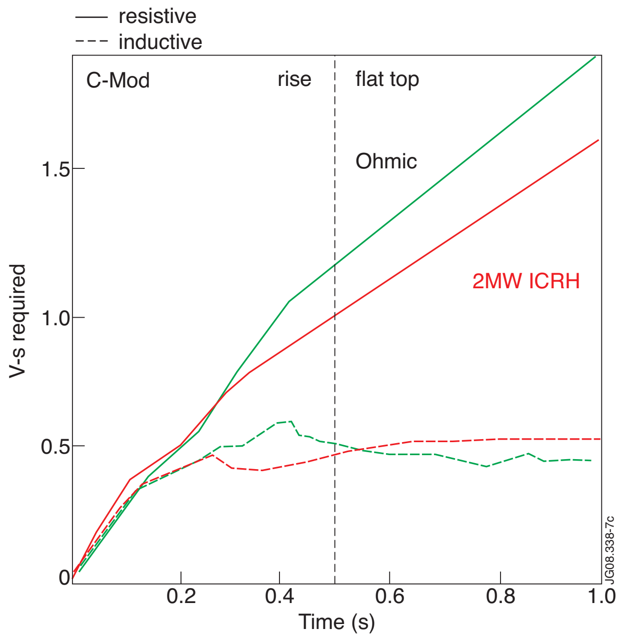


Figure 6*: Heated current rise phases in ASDEX Upgrade (a) and JET (b). A reduction of $l_i(3)$ with heating during the current rise is observed. For ASDEX Upgrade a current rise of dI_p/dt=0.66MA/s is used with up to 2.5MW of NBI heating in L-mode and H-mode. For JET a current rise with dI_p/dt=0.28MA/s is used, experiments in L-mode with moderate heating (3-5MW) usually achieve $l_i(3)$~0.85, indicated by the blue dotted line for reference. For JET, transitions to H-mode and phases with an Internal Transport Barrier (ITB) are indicated.*



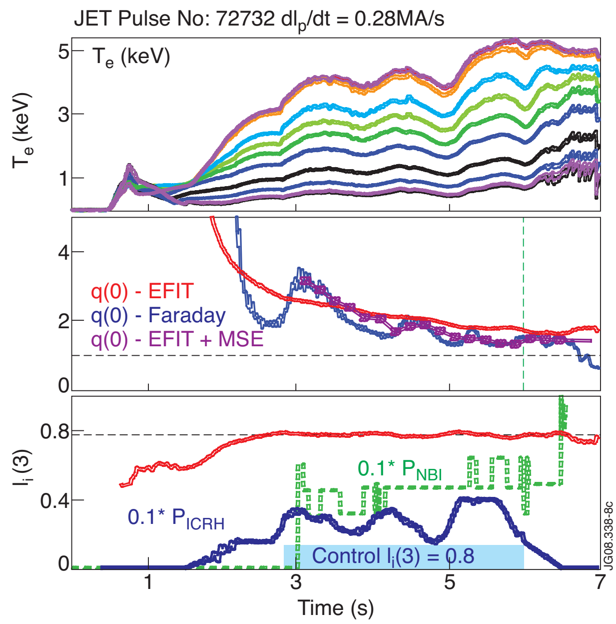


Figure 7: Interpretation of the current rise results from C-Mod using TRANSP/TSC [2]. An ohmic discharge (green) and a 2MW ICRF heated discharge (red) are compared. Given are the resistive flux consumption (solid lines) and the inductive flux consumption using the Poynting method to calculate the consumed flux by the plasma [2].

Figure 8: A JET discharge ramping to q_{95} = 4 at dI_p/dt=0.28MA/s. Shown is a control of l_i(3)=0.8 with ICRH, giving q(0)>1 at 6s. The level evolution of NBI power to give same l_i(3)=0.8 (previous pulses) is indicated. Shown from top to bottom are the electron temperature at various radii, the evolution of the q(0) using different reconstruction methods, and the l_i(3) evolution compared to the set point of 0.8, together with the ICRH and NBI power used (note, power is in 10MW units).

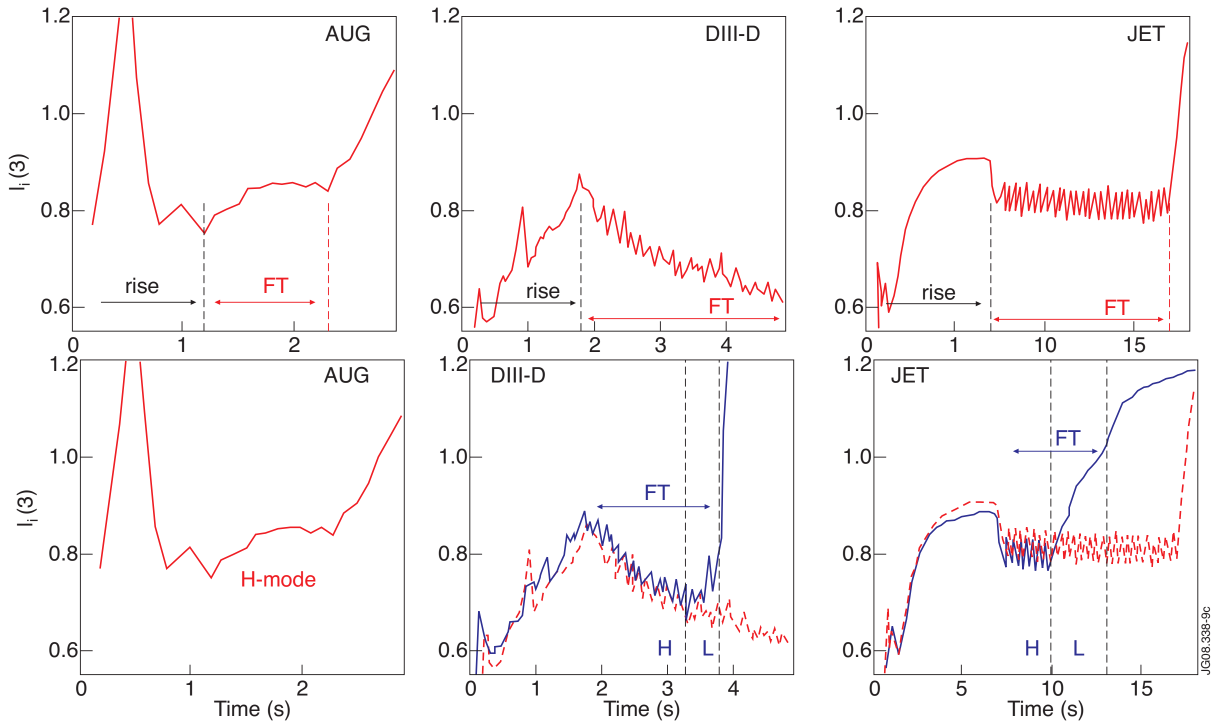


***Figure 9:** The l_i(3) evolution for ITER demonstration discharges at q_{95}=3. From left to right are shown data from ASDEX Upgrade, DIII-D and JET. All discharges enter H-mode at the start of the flat top. (a) Both current rise and flat top phase are shown. (b) The discharges indicated in blue (for DIII-D and JET) have a deliberate step down of the heating power during the flat top phase to provoke a transition back to L-mode (at 3.5s for DIII-D and at 10s for JET).*

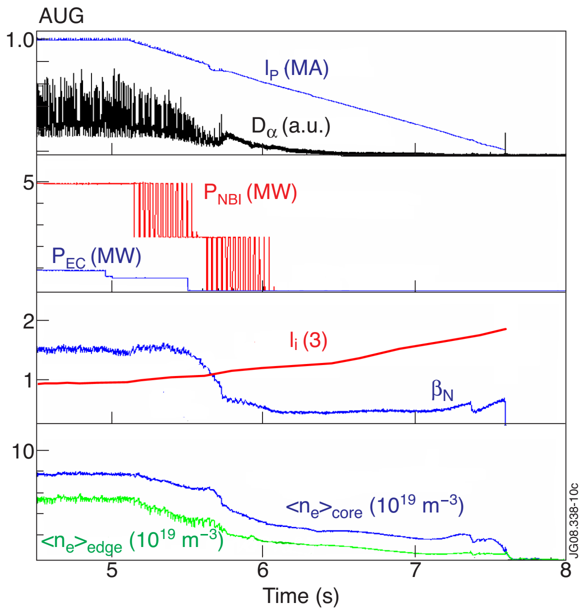


Figure 10: Current ramp down experiments in ASDEX Upgrade as an example of an ITER like ramp down phase. Show from top to bottom are the plasma current overlaid with the D_α emission measurements, the neutral beam power and ECRH power used, the evolution of li(3), combined with the evolution of β_N, and finally the evolution two line averaged density measurements; one chord going through the entire plasma and one going only through the edge of the plasma.

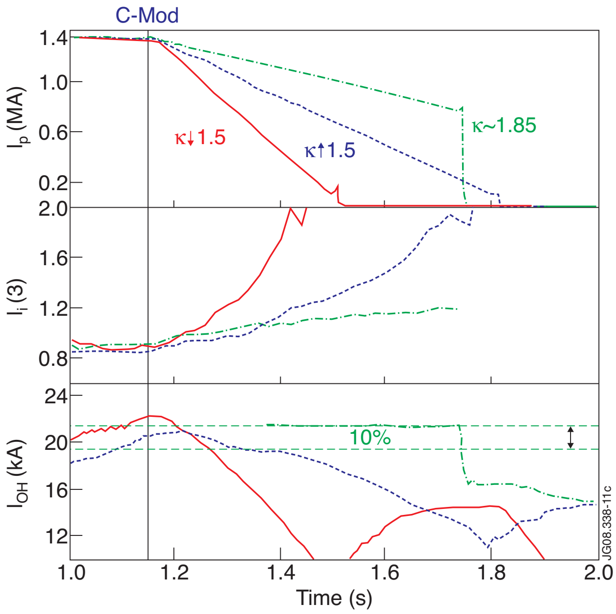


Figure 11: Current ramp down experiments in C-Mod, varying the Ip ramp down rate from 4MA/s (red curves) to 2MA/s (blue curves) and 1MA/s (green curves). Indicated is the reduction of the elongation (κ) during the current rise phase. Shown from top to bottom are the plasma current, the evolution of the plasma inductance li(3) and the current in the central part of the OH transformer in C-Mod.