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Alcator C-Mod Experiments in Support of the ITER Baseline 15 MA Scenario

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Abstract. Experiments on Alcator C-Mod from 2009-2012 have examined the rampup, flattop and rampdown phases of the proposed ITER 15 MA baseline scenario. Rampup studies show ICRF heating can significantly reduce the V-s requirement, and that an H-mode late in the ramp can reduce this further. ICRF modifications to l_i in L-mode are minimal, although the Te profile is peaked relative to ohmic in the plasma center, and reduces the sawtooth onset times. Flattop plasmas targeting ITER baseline parameters have been sustained for 20 τ_E or 8-13 τ_{CR}, but only reach H_{98} ~ 0.6 at n/n_{Gr} = 0.85, rising to 0.9 at n/n_{Gr} = 0.65. Rampdown studies show H-modes can be routinely sustained with ICRF power injection, avoiding an OH coil over-current associated with the H-L transition. In addition, faster current rampdowns are preferred to avoid an over-current when an H-L transition ultimately does occur. In the H-mode rampdown the density is found to drop with Ip, preserving the n/n_{Gr} ratio, so long as ICRF power is injected.

1. Introduction

Alcator C-Mod [1] has performed a number of experiments from 2009-2012 to examine issues associated with the baseline 15 MA ELMy H-mode scenario in ITER. Numerical simulations of this scenario [2], which were done to confirm that sufficient operating space existed, identified a number of uncertain features that would benefit from experimental support. In the plasma current rampup phase the volt-second requirement and resulting current profile (characterized by the internal self-inductance, $l_i(1)$, (defined below) depends on the ramp rate, level of ohmic and auxiliary heating, type of auxiliary heating, global energy confinement, L or H-mode regime, and level of impurities. Experiments were done to examine ohmic and ICRF heated rampup, with a rampup time characteristic of ITER, t_{ramp}/τ_{CR} ~ 10 (τ_{CR} is current redistribution time). Simulation of these experimental rampup phases allows the comparison of energy transport models, which are typically used to project ITER behavior, with the actual experimental evolution, and will be reported separately. In the experiments examining the flattop phase, the q_{95}, plasma shape, β_N, n/n_{Gr} (n_{Gr} = $I_P/\pi a^2$), and H_{98} are targeted to match those anticipated in ITER, and the demonstration of this regime with multiple τ_{CR} duration in C-Mod has been accomplished. In the rampdown phase, numerical simulations showed [2] that allowing an H-L transition at the end of flattop burn in ITER would result in an over-current in the central solenoid (CS) coils as these coils compensate the sudden change in poloidal flux. Although remaining in H-mode into the rampdown was shown to eliminate this effect, additional questions remained for this discharge phase; what is required to stay in the H-mode, plasma shape evolution required to avoid vertical instability, how did the density evolve, and could the plasma remain diverted to low plasma current values avoiding excessive heating of the first wall tiles. The activities on C-Mod to address these issues are reported here.

The term "volt-seconds" refers to the poloidal magnetic flux swing, provided by the poloidal field coils, required to establish the plasma geometry, current profile, and total plasma current. Here we will define the phrase volt-second requirement, abbreviated by V-s-R. This primarily refers to the current rampup phase of the discharge. It is composed of an external inductive contribution to establish the magnetic field outside the plasma, an internal inductive contribution to establish the magnetic field inside the plasma, and the resistive contribution, which is consumed through ohmic dissipation. A simplified expression for this can be given by

$$\Delta\psi = \Delta\psi_{ind,ext} + \Delta\psi_{ind,int} + \Delta\psi_{res} = \mu_o I_p R\left(L_{ext} + \frac{l_i}{2} + C_E\right) = \psi(R, Z, t_{EOR}) - \psi(R, Z, 0)$$

Here Ip is the plasma current, R the major radius, L_{ext} is the external self-inductance [3], l_i is the internal self-inductance (expression used here is $l_i(1) = \int B_\theta^2\, dV/[\{(\mu_o I_p)/\int dl_\theta\}\times 2\pi R \int dA_\phi]$), and C_E is the Ejima coefficient [4] representing the resistive flux swing. The $\psi(R,Z,t)$ is the poloidal magnetic flux linked through the reference location (R,Z) from all poloidal field coils (ohmic and equilibrium), and t_{EOR} is the end of current rampup time. This linked poloidal flux can be defined with different reference locations [2], however the differences between these are small, and for the discussion here this reference can be considered the plasma magnetic axis. The Ejima coefficient is just a convenient factor to represent the resistive consumption in a similar way as the inductive terms. There are two accounting methods for poloidal magnetic flux, the Axial and the Poynting methods [5], based respectively on flux balance and energy balance, and the expressions for each term can be found in that reference. The resistive volt-seconds are lost as heat in the plasma, while in principle the inductive volt-seconds can be recovered in current rampdown, but practically this is not achieved. The flattop plasma, if the plasma current is fixed and the current profile is relaxed and unchanging, will only require resistive volt-seconds to sustain the plasma current, and this would add another term to the expression, $\Delta\psi_{res,flattop}$. We will examine the poloidal magnetic flux change at the plasma edge during the current rampup, derived from equilibrium reconstructions, which represents the internal volt-seconds requirement, composed of the internal inductive and resistive terms. The external inductive term is the same for all rampup plasmas since the geometry and the total plasma current are made as similar as possible.

Alcator C-Mod is a high field compact tokamak, with a major radius of 0.67 m, minor radius of 0.22 m, elongation of up to 1.8 and triangularity up to 0.5, and a standard toroidal field of 5.4 T. These experiments utilize first harmonic (ω_{cH}) minority hydrogen ion cyclotron (IC) heating, in a deuterium background plasma, at 80 MHz, and a corresponding plasma current of 1.3 MA (for $q_{95} \sim 3$). Some of the experiments reported here are at 2.7 T and 650 kA, and 2nd harmonic ($2\omega_{cH}$) minority hydrogen heating, in order to access both the flattop target β_N and n/n_{Gr} simultaneously. The L to H transition thresholds [3] for flattop plasmas are approximately $P_{thr,LH}$ (MW) = $1.5 n_L^{0.72}$ for 5.4 T and $0.86 n_L^{0.72}$ for 2.7 T, for line average density in 10^{20} /m^3. Fig. 1 shows a cross-section of the Alcator C-Mod tokamak, with ohmic (OH) and poloidal field (PF) coils, and various limiting conducting structures. For all plasmas shown here the ICRF was in dipole phasing and did not drive fast wave current.

1. Plasma Current Rampup

The anticipated ITER current rampup time can range from ~ 50 to 100 s, while a range of the current redistribution time (τ_{CR}) is about 5-16 s, showing that the plasma current profile will be largely relaxed and that the current density profile will directly reflect the electron temperature profile, that is the surface averaged parallel current density $<j_{||}>$ is proportional to $T_e^{3/2}$. In C-Mod the rampup time was 450-550 ms, with a corresponding τ_{CR} range of 25-55 ms for ohmic and ICRF heated discharges. C-Mod's main heating source is up to 5 MW of ICRF, while 10 MW of ICRF power is planned for the day-one heating mix on ITER, with possible expansion to 20 MW with an additional antenna. A comparison of the ohmic rampup and ICRF heated rampup was pursued to define reduction of the V-s-R associated with heating and how this heating method would modify the current profile. The plasmas are grown to full size and diverted in the 60-100 ms time frame, corresponding to the ~15 s divert time (out of 50-100 s rampup time) in ITER. This can significantly reduce Z_{eff} early in the ramp phase, and reduce the variability of the rampup phases. The toroidal field is 5.4 T, plasma current is 1.3 MA, and the elongation is 1.75. The end of rampup target line average density is 1.35×10^{20} /m^3, giving n/n_{Gr} = 0.16, and a volume average collisionality of 0.08 with ICRF heating and 0.15 for ohmic discharges.

Shown in Fig. 2 are the OH1 coil currents, internal self-inductance (li), line average density (/m^2), and electron central temperature versus time, for ohmic heating only, with 1 MW of ICRF heating injected from 120 to 500 ms, with 1 MW from 120 to 250 ms and 2 MW from 250 to 500 ms (1-2 MW) of ICRF heating, and the same with an H-mode transition at 410 ms out of 500 ms of the rampup time. The ICRF power was initiated at 120 ms, and always began with 1 MW, since higher powers resulted in larger impurity influx and radiated powers. The lower OH1 coil current at the end of the ramp for cases with ICRF heating, show that fewer V-s are required to rampup the plasma current. The difference in OH1 coil current between injecting 1 MW and 1-2 MW in L-mode is insignificant in spite of higher central electron temperature, but also higher radiated power, in the 1-2 MW case. The onset of H-mode during the rampup allows even larger V-s saving, and although not shown, may allow higher heating power to be effective. A discharge with an H-mode forming at the same time as the case shown in Fig. 2, also had the ICRF power step up from 2 to 3 MW at the same time, and demonstrated further reduction of the OH1 current and the edge poloidal flux, however, it was not sustained to end of rampup. Fig. 2 also shows the radiated power, plasma elongation, ICRF power, and the plasma edge poloidal magnetic flux. The radiated power is typically higher in ICRF heated L-mode rampups than in ohmic cases, and tends to show a strong increase in the later phase of the rampup. The edge poloidal flux reflects the resistive and internal inductive flux required to form and sustain the plasma. The relative V-s savings derived from the plasma edge poloidal flux is 9.6% of the ohmic V-s for the 1 and 1-2 MW ICRF heated cases in L-mode, while the savings are 14.4% of the ohmic V-s for the 1-2 MW with an H-mode. The experimental discharges demonstrate some variability in V-s and OH1 coil current values at the end of the current rampup, in spite of nominally similar parameters. This arises due to slightly different density trajectories, Z_{eff} behaviour, and wall conditions. Comparing multiple ohmic and multiple 1-2 MW ICRF heated cases, the variations among ohmic rampups amount to 5.4% of a nominal ohmic swing, and among ICRF heated rampups to 2.4% of a nominal 1-2 MW ICRF heated swing, indicating the observed savings exceed the natural discharge to discharge variation by a sufficient amount. The plasma edge poloidal flux swing and the OH1 coil current at the end of the ramp are shown in

Fig. 3 for the 5 discharge types discussed above, showing the reduction with heating and H-mode onset compared to ohmic. The fifth case with an H-mode is a projection of the V-s if the H-mode had been sustained.

Saving V-s in ITER is desirable in order to guarantee a sufficiently long flattop burn phase. The OH1 coil in C-Mod is very similar in behavior to the centermost central solenoid (CS) coil in ITER, since it provides the full swing from negative maximum current through zero current and to the positive maximum current, predominantly driving inductive plasma current. This current swing behaviour is what is seen in simulations of the ITER baseline discharge with the proposed segmented CS. Using these savings fractions, derived from C-Mod discharges, relative to an ohmic ITER rampup, which requires about 115 V-s in 100 s [2], the savings would be up to 11 V-s for heated L-mode, 17 V-s for the heated L-mode with late ramp H-mode, and 23 V-s for late H-mode with additional ICRF heating. The OH1 coil current savings amount to 5.6% and 9.7% of the ohmic current swing. The corresponding OH1 current swings represent the entire V-s requirement, including the external inductive contribution, which is considered very similar in all cases due to the same plasma current and shape. Comparing these fractions to ITER, which requires a central solenoid (CS1) coil swing of about 85 MA (double swing) for an ohmic rampup in 100 s, the savings correspond to 4.76 and 8.25 MA.

The plot of li in Fig. 2 shows little change from ohmic at the end of the ramp for L-mode when adding ICRF heating in the rampup phase. Since the current relaxation time is short compared to the rampup time, we expect the current profile to be relaxed, and therefore have a shape that is proportional to $T_e^{3/2}$. Fig. 4 shows four electron temperature profiles from Thomson scattering and ECE measurements at 177, 227, 294 and 477 ms during rampup, for the cases being considered. The peaking of the temperature profile in the center is clearly demonstrated, relative to a weaker change in the outer half of the minor radius. This T_e profile change will not contribute strongly to the integrated quantity li (proportional to $\int B^2_{pol} dV$), since both the poloidal field and volumes are small in the plasma center. On the other hand, the sawtooth onset time, which is a sensitive measure of the on-axis current density, does show the effect of ICRF and corresponding T_e profile peaking, which causes earlier onset than for an ohmic plasma. The sawtooth onset times are 350 ms for the ohmic discharge, and 240, 250, and 300 ms for the ICRF heated discharges, shown in by the central electron temperatures in Fig. 2. The ICRF power deposition profiles onto thermal ions and electrons are shown Fig. 5, along with the total power split between electrons and ions, calculated from TORIC full wave analysis [4] in TRANSP. Here the hydrogen minority absorbs nearly 100% of the injected power due to very strong damping, so that the heating on thermal electrons and ions is entirely from slowing down. The deposition is within the normalized minor radius of 0.3, reflecting where the strongest modification of the T_e profile occurs compared to the ohmic profile. Therefore, the current profile is not modified using a measure like li(1), but is locally modified in the plasma center.

From these observations of L-mode ICRF heated rampups, the V-s savings over an ohmic rampup are determined to be resistive, and any internal inductive contribution, associated with the L-mode current profile (li) evolution, was found to be minimal. The resistive V-s savings are due to higher electron temperatures overall in the ICRF heated plasmas, in spite of net higher core radiated power losses with ICRF heating. Following the OH1 current and edge poloidal magnetic flux time histories through the flattop phase, where the ICRF heated plasma enters and exits the H-mode, then the V-s saved, by heating with ICRF in rampup, can be shown to be preserved to the end of flattop. The rate of change of the OH1 coil current will slow down as the plasma enters H-mode and speed up when it exits H-mode at the end of the flattop, and this behavior is similar for the edge poloidal magnetic flux. These are due to a change in the current profile as the pedestal generates bootstrap current, lowering li, and producing an inductive V-s savings. The V-s reduction at H-mode onset is subsequently consumed at the H-mode exit, and would not typically be considered as savings. However, as will be discussed later, preserving the H-mode into the rampdown is in fact the reference for ITER. H-mode preservation into rampdown has been routinely demonstrated on C-Mod, and the corresponding V-s savings associated with H-mode onset in the rampup can then be counted as savings.

Global energy confinement time scalings relative to IPB98(y,2) obtained in these rampup discharges are H_{98} = 0.35-0.4 for ohmic discharges, and 0.45-0.6 for ICRF heated discharges. Radiated powers from the core plasma during rampup ranged from about 0.2-0.25 MW for ohmic ramps, to 0.25-0.5 MW with ICRF heating in L-mode and H-mode, and up to 1.0-1.5 MW for early high ICRF power injection. The ICRF heated rampups had their radiated power increase significantly in the last 20% of the ramp, increasing from 0.25-0.5 MW to 1.0-1.5 MW, due to an influx of impurities, since the densities were constant during that time, and the temperatures are unaffected. The impurity source may be associated with the cross-over in the OH1 coil current (when it passes through zero), however the connection with an impurity source has not been established. Radiated power fractions relative to the input power during the rampup phase ranged from 0.2-0.5 for ohmic discharges, from 0.12-0.3 for ICRF heated discharges, and from 0.07-0.63 for ICRF heated discharges in the last 20% of the rampup. The power thresholds for L to H transition, for the two cases with H-modes in the rampup, are $P_{loss}/P_{thr,LH}$ = 1.21 and 1.61, and $P_{net}/P_{thr,LH}$ = 0.9 and 1.15. P_{loss} is defined as $0.9P_{ICRF} + P_{OH} - dW/dt$, and $P_{net} = 0.9P_{ICRF} + P_{OH} - P_{rad} - dW/dt$, with the 0.9 factor to account for losses of ICRF power. The discharge 024, shown in Fig. 2, which transitioned into H-mode in spite of having the same ICRF power (about 2 MW), as 018 at that time, had a core radiated power of 0.65 MW, and an ohmic power of 0.6 MW. While 018, which remained in L-mode, had a core radiated power of 0.85 MW, and an ohmic power of 0.85 MW. The reason that 024 transitioned while 018 did not is not clear, however, in general H-modes in the current ramp are difficult to produce in C-Mod. Although radiated power is not typically included in L-H threshold evaluations, net power may be a more appropriate measure when radiated powers are a significant fraction of the input power. The sudden drop in ICRF power about the same time as the L-H transition is actually a response to the sudden change in edge plasma density affecting the antenna coupling.

The impact of density variations in the rampup phase was examined by scanning the final density at end of ramp through 0.6, 1.0, 1.35, and 1.8×10^{20} /m^3. These correspond to a range of densities relative to the Greenwald density of 0.07 to 0.22. Considering the ohmic rampup cases, although the decrease in density showed a complimentary increase in electron temperature (approximately constant pressure), the reduction in OH1 coil current or edge poloidal magnetic flux is minimal. The lowest density resulted in locked mode disruptions in every case, indicating this is below a safe operating density. The total radiated powers among the discharges are similar, and the edge poloidal magnetic flux shows only a weak variation. The current profiles, as described by li, do not show significant differences at the end of ramp, and the sawtooth onsets, from ECE central temperature, are earlier for the lower temperatures, as would be expected from current diffusion. The lowering of plasma density in the rampup phase does not appear to reduce V-s consumption or modify the current profile in any significant way, apart from very low densities where the locked mode limit is reached at $n/n_{Gr} \geq 0.07$. For ICRF heated discharges, with 1.35 and 1.8 x 10^{20} /m^3 end of rampup densities, no significant change in OH1 coil current or edge poloidal magnetic flux was observed.

2. Flattop Plasmas

The baseline flattop burning plasma in ITER targets simultaneous parameters $q_{95} = 3$, elongation of 1.8, $\beta_N = 1.75$, $n/n_{Gr} = 0.85$, and $H_{98} \sim 1.0$, while in a regulated H-mode (i.e ELMy H-mode). C-Mod discharges have attempted to reproduce this set of parameters by reducing the toroidal field from a typical value of 5.4 T, to 2.7 T, which allows higher β_N and n/n_{Gr} simultaneously. The plasma current is 0.65 MA, elongation is 1.75, q_{95} reaches 3.0-3.4, and 2nd harmonic ($2\omega_{cH}$) hydrogen minority heating is used at 80 MHz. The rampup and rampdown times for these discharges are shortened to reflect a reduction in the current diffusion time due to lower electron temperature (250-300 ms). ITER must also radiate a significant portion of its input power to maintain the power to the divertor within acceptable levels, which ranges from 25-40%. C-Mod has a metallic wall made of molybdenum, and a boron low Z coating is applied. The fractions of radiated to input power ($P_{ICRF} + P_{OH}$) from these intrinsic impurities ranged from 15-45%, quite close to the ITER expectations. This radiated power ratio tends to increase with increasing density, however there is significant spread in the data, for example 22-42% at a line integrated density of 3.5x10^{20} /m^2. In addition, this ratio tends to slightly decrease or remain similar at higher ICRF powers. The H-mode is regulated by a quasi-coherent mode (QCM) and is in the enhanced D-alpha regime (EDA). The ICRF heating waveform was varied to minimize or avoid the ELM-free phase, although this was not completely successful, and several discharges show some ELM-free features in the early part of the high power flattop phase (3-5 MW injected ICRF power). Shown in Fig. 6 are waveforms for 3 discharges typical of the parameter space obtained. One shows an n/n_{Gr} ratio of 0.85 with β_N reaching 1.35, the next obtains n/n_{Gr} of 0.75 with β_N = 1.75, and another with n/n_{Gr} of 0.65 and $\beta_N > 2.0$. The associated H_{98} factors were 0.55 for n/n_{Gr} = 0.85, 0.72 for n/n_{Gr} = 0.75, and 0.83 for n/n_{Gr} = 0.65. Quasi-stationary phases are sustained for ~ 400 ms, equivalent to 20 energy confinement times (τ_E = 20 ms), or 8-13 current redistribution times (τ_{CR} = 30-50 ms).

The H-mode onset powers for these flattop discharges experiments obtained a range of ratios for $P_{loss}/P_{thr,LH}$ = 1.4-2.35 (with the majority of the cases between 1.85-2.35) and $P_{net}/P_{thr,LH}$ = 1.15-2.0 (majority between 1.4-2.0), respectively. Ohmic H-mode onsets occurred in a few discharges with loss power ratios of only 0.4-0.8, and net power ratios of 0.2-0.55. These powers are shown in Fig. 7 for 2.7 T and 5.4 T experiments. As mentioned earlier, the threshold formula [3] does not represent the actual threshold powers in C-Mod as a function of the toroidal field and plasma density, and is shown in the figure. ITER is expected to operate in flattop near the L-H threshold power with $P_{loss}/P_{thr,LH}$ of ~ 1.7 or $P_{net}/P_{thr,LH}$ of ~ 1.25 depending on the radiated power levels. The corresponding ratios of loss and net power in the C-Mod flattops were 1.25-2.75 and 0.8-2.20, respectively. During flattop the ohmic heating power in ITER will be very small, approximately 1-2 MW, giving $(P_{AUX} + P_{OH}) / P_{OH} > 60$ in the flattop phase, and the C-Mod experiments reach a range of 2.5-13 in flattop. The heating power deposited on the electron thermal channel in the ITER baseline, including fusion alphas, results in a $P_{e,th}/P_{input} \sim 64\text{-}72\%$, with its precise value depending on the auxiliary heating method. The C-Mod experiments at 2.7 T reached values of this ratio 81-88%, while at 5.4 T this ratio is about 66%, and both of these include the ohmic power contribution on the electrons.

Examining the quasi-stationary phase of the discharge, the global energy confinement was strongly reduced as the higher densities are achieved. A series of parameter comparisons for both 2011 and 2012 flattop experiments are shown in Fig. 8. In 2012 the goal was to increase the net power in conjunction with reaching high density (high n/n_{Gr}) in order to increase the energy confinement, as was indicated by data both from separate experiments [6] and the ITER flattop experiments performed in 2011. Although P_{net} was increased the improvement in energy confinement was limited. The net power normalized to the L-H threshold power was increased in the 2012 experiments beyond the 2011 cases up to 1.5-2x the threshold, but in spite of this, the confinement remained similar or only slightly higher than the older experiments. The stored energy in the plasma is observed to decrease with n/n_{Gr}, and the net power delivered to the plasma is also decreasing with n/n_{Gr}. C-Mod does not have an active density control capability for H-mode plasmas, and the resulting density in the flattop phase can vary. Comparing two discharges with the same ICRF and net powers (2.8 and 2.1, respectively), with high and low n/n_{Gr} (0.84 and 0.68), it is found that the approximate 25% increase in density has resulted in ~2.5x decrease in the temperature at the pedestal and a ~25% reduction in the plasma core. The pressure is not preserved in this trade-off, and the pedestal temperature is severely reduced. The electron density and temperature profiles are shown in Fig. 9. The resulting global confinement multiplier H_{98} dropped from 0.8 to 0.53. The cooling of the plasma edge may be attributed to resistive ballooning like modes as identified in numerical studies of the approach to the Greenwald density limit [7]. In these studies, the turbulence in the scrape-off layer penetrates the separatrix and cools the plasma edge region. It is possible that the strong edge fueling or reduced heating efficiency of 2nd harmonic ICRH at higher densities could contribute to this trend as the density approaches the Greenwald density. Previous work in JET discharges [8] showed that higher plasma triangularity could restore the global energy confinement at high n/n_{Gr}. For C-Mod the upper triangularity is in the range of 0.38-0.42, and lower triangularity (X-point) of 0.5-0.54. Overall the average triangularity is ~ 0.46, which is similar to the values obtained in JET, however, a confinement recovery has not been observed. More dedicated experiments modifying both the upper and the lower triangularity are needed to explore if this is accessible.

Strong MHD activity ($B_r/B_\theta \sim 10^{-3}$) is observed in many discharges. The most persistent modes have frequencies in the 7 to 12 kHz range, with toroidal mode number n=2 and rotating in the co-current (ion diamagnetic) direction in the lab frame. There are also modes at ~ 15-20 kHz, with toroidal mode numbers of 3 and 4. The core plasma rotation is found to have speeds of ~ 50 km/s in the co-Ip direction, consistent with other H-modes in C-Mod [9], whether ICRF heated or ohmic, scaling as ~ 400 W_{MHD}/Ip. The magnetic signature of the n=2 mode is consistent with a poloidal mode number m=3, but soft X-ray arrays indicate a complicated radial structure with coherent components in the pedestal region and near the q=1 surface, as well as near q=3/2. The mode onset can be triggered by a transient event, most often an H-L-H transition (due to an ELM-free phase) with sharp drops and recoveries in β_N and ICRF power, as well as a sharp rise and fall in li. The mode onset is observed with this transient when $1.1 < \beta_N < 1.35$. Mode onset also occurs spontaneously as β_N rises above 1.5. Shown in Fig. 10 is the value of β_N at mode onset against the plasma internal self-inductance li(1), for 10 kHz (X) and 15-20 kHz (Ø), showing the clear separation between those that occur due to transients at lower β_N and higher li (red), and those that are spontaneous at higher β_N and lower li (green). Also shown are the discharges, which did not establish a mode, which appear to fall in between, although there is some overlap with the spontaneous onset region. For these discharges the β_N and li reported are the values reached in the quasi-stationary phase. The transient onset cases are exclusively the ~ 10 kHz type, in fact, the one shown as a circle (Ø) in the transient onset zone actually starts out at 10 kHz and evolves to the higher frequency. The spontaneous onset has both the lower and higher frequency modes. Once initiated, the modes persist despite small variations in β_N, but the frequency rises with decreasing β_N and vice versa. The modes often persist into rampdown as long as the H-mode is sustained, terminating when β_N drops below 1.0 to 1.35. The degree to which these modes may degrade global confinement is not clear, but the mode amplitudes do not seem to be correlated with n/n_{Gr}. Examining the onset for neo-classical tearing modes, from refs [10,11], with the appropriate $\rho_{i\phi}$ and ν_{e*} values for these discharges, gives β_N of 3.85 and 3.75, well above the values reached in these discharges. So the modes are expected to be classical tearing modes. No significant correlation of the mode presence with a reduction of global energy confinement can be found.

3. Plasma Current Rampdown

The rampdown phase in ITER has a complex combination of constraints, 1) reduce the plasma current to < 15% of its flattop value, 2) avoid vertical instabilities, 3) stop consuming V-s (or advancing the CS coil currents), 4) remain diverted, and 5) avoid disruptions of all types until the plasma has very low current. It was found in simulations of ITER that the sudden exit from H to L-mode would induce a response from the CS coils, referred to as an over-current. This is undesirable at the end of flattop in ITER because ITER will be trying to push coils near their limiting currents and fields to maximize the flattop burn duration. This over-current can be avoided by remaining in the H-mode as the discharge transitions from flattop into rampdown, and subsequently maintaining the H-mode for some period of time. However, it was generally unknown what the plasma density would do in the rampdown phase as the plasma current was reduced, particularly with the aggravating features of high density relative to Greenwald and seeded impurities. The time scale of the rampdown for ITER was unspecified as there was little guidance on the requirements.

For these experiments, an EDA H-mode was established before plasma current rampdown, by injecting 1-3 MW of ICRF power during the flattop phase. Three rampdown times were examined, 500 ms, 750 ms, and 1000 ms, corresponding to approximately 120, 180 and 240 s in ITER based on current diffusion times. The experiments were at a toroidal field of 5.4 T, flattop plasma current of 1.3 MA to obtain $q_{95} \sim 3$, and begin rampdown with line average densities ranging from $4.8\text{-}5.2\times10^{20}$ /m^3 giving n/n_{Gr} = 0.54-0.58. The plasma elongation was reduced with the plasma current, from 1.75 to 1.4-1.5, in order to avoid vertical instability that could arise with evolution to higher li, more peaked current distributions. The plasmas remained vertically stable even with li rising to greater than 2.0. The plasma is maintained on the midplane (Z = 0) during the rampdown in order to allow ICRF power coupling to the plasma. In general, it was possible to bring the plasma current down to 15% of its flattop value before termination, although some cases suffered from injections (very rapid radiative collapse), which terminated the discharges, however none were terminated due to vertical instability or progressive impurity accumulation (and associated rise in radiated power). The sustainment of the EDA H-mode in the rampdown required ICRF power injection, generally $\geq$ 0.75 MW. The experiments showed that the plasma density decreases at the same rate as the plasma current, and therefore maintained a constant n/n_{Gr}, when in a regulated H-mode (such as an EDA), which is only sustained with ICRF power. When no ICRF power was applied in the rampdown, the plasma density does not drop with Ip, but can remain high followed by an H-L transition. Shown in Fig. 11 are plots of the plasma current, OH1 current, density, and ICRF power for cases maintaining regulated H-modes into the rampdown phase. Also shown are the li(1), q95, plasma elongation, and plasma stored energy. The plasma stored energy decreases approximately with the square of the plasma current. The slowest rampdown does not obtain a decrease in the OH1 coil current, making this case vulnerable to the over-current, which is evident at 1.85 s where the OH1 coil current rises with the H-L transition. The other faster rampdown cases obtain a drop in OH1 coil current, and the response to the H-L transition is barely visible. In ITER the precise time when an H-L transition could be tolerated will depend on several variables including the termination of auxiliary power since the plasma would be moved downward away from the midplane, and toward the divertor, and when the central solenoid coils were sufficiently low that the over-current can be tolerated. Here it is demonstrated, in addition to sustaining the H-mode into the rampdown phase, that faster rampdowns are preferable since they provide the largest margin for H-L transitions, intentional or unintentional, since the OH1 coil is reduced more quickly.

The pedestal in the H-mode during the rampdown, shows that both temperature and density are decreasing in Fig. 12, however, not at the same rate. The density does follow the Ip, while the temperature tends to drop rapidly early and then very slowly. This temperature response is consistent with the drop in the ICRF power when transitioning from the flattop to rampdown, with the heating in the plasma center and the pedestal near the plasma edge. The global energy confinement time during this phase is ~ 45 ms, with a slowing down time of the hydrogen minority of about 12 ms. Eventually H-L transitions do occur in spite of continued injected power. For the regulated H-modes, over all the rampdown rates, the back transitions occur at ratios of $P_{loss}/P_{thr,LH}$ over the range of 0.57-0.68 and $P_{net}/P_{thr,LH}$ over the range of 0.37-0.46, indicating a hysteresis or some benefit associated with ramping the plasma current down. During these rampdown H-modes the energy confinement obtained ranges over H_{98} ~ 0.6-0.67 in the beginning of rampdown to values ~ 0.3-0.47 just prior to the H-L back transition. During the rampdown phase the ohmic heating power ranges from 2.0 MW at the earlier times to 0.7-1.7 MW at the H-L back transition, compared to 0.75-1.3 MW of ICRF injected power. This ohmic power contributes strongly to the total input power to the plasma. In ITER the ohmic heating power would not exceed ~ 10 MW (compared to 73 MW of auxiliary heating available), while the L-H threshold formula for full shaping gives ~ 90 $n^{0.72}$ MW [3], and so establishing and maintaining an H-mode in rampdown in ITER must rely almost entirely on auxiliary power, and any residual alpha power in the early higher density phase.

Flattop experiments (Ip = 0.65 MA, B_T = 2.7 T), discussed earlier, that reach n/n_{Gr} ratios of 0.75-0.89 at the beginning of rampdown all showed sustained H-modes into the rampdown with 1.6 MW of injected power, dropping to 1.1 MW late in the rampdown. The density tracked the plasma current as seen before. These cases also avoided OH1 coil over-currents, and ramped down to 15% of the flattop plasma current before terminating. These required power levels were higher than those required for the lower n/n_{Gr} discharges obtained with Ip = 1.3 MA and B_T = 5.4 T, indicating that the higher relative densities might require greater power to sustain the H-mode. The H-L back transitions occurred at ratios of $P_{loss}/P_{thr,LH}$ in the range 1.06-1.33, and $P_{net}/P_{thr,LH}$ in the range of 0.70-0.90, noticeably higher than those at lower n/n_{Gr} for B_T = 5.4 T. The higher ratio may be misleading, as has been pointed out in [5]. The threshold power in C-Mod is found to be similar over a range of toroidal fields, over a wide range in density, and even as the density rises from the minimum. On the other hand, the threshold formula indicates a strong B_T dependence, which gives rise to an apparent increase in the power ratio $P_{loss}/P_{thr,LH}$. In these experiments the ohmic power at the back transition is in the range of 0.25-0.55 MW, compared to injected ICRF powers of 1.1-1.6 MW, moving much closer to the ITER situation. Dedicated rampdown experiments at 2.7 T will be necessary to more precisely quantify the power requirements to sustain H-mode.

4. Conclusions

Experiments on C-Mod in support of the ITER 15 MA ELMy H-mode baseline scenario have clarified several important issues raised in the recent design studies and as part of the ITPA Integrated Operating Scenario activities [12]. The application of ICRF power has been shown to reduce the V-s-R by 10% in the rampup phase for L-mode compared to ohmic rampup, and about 15% if an H-mode is induced late in the ramp. For the L-mode cases the V-s savings is determined to be resistive, due to higher electron temperatures, with no inductive contribution. Changes of li(1) from the ICRF heating in L-mode are minimal compared to ohmic current profiles, due to the specific heating deposition and Te profile response. The inductive V-s savings associated with an H-mode current profile modification can be credited since the plasma remains in H-mode sufficiently far into the rampdown. Global energy confinement in the rampup phase are H_{98} = 0.35-0.4 for ohmic and 0.45-0.6 for ICRF heated discharges. Density variations during the rampup did not demonstrate V-s savings unless it was low enough to induce locked mode disruptions. Flattop experiments at reduced toroidal field of 2.7 T, targeting the ITER baseline parameter set of q_{95}, shape, β_N, n/n_{Gr}, and H_{98}, showed that the $H_{98} \sim 1$ could not be reached in combination with the high n/n_{Gr} of 0.85, although it was recovered when n/n_{Gr} decreased to 0.65. These discharges showed quasi-stationary phases at high performance for 20 τ_E and 8-13 τ_{CR}. Simultaneously, EDA regulated H-modes were obtained, impurities were low Z boron and high Z molybdenum, and radiated power fractions of 15-45% were reached. MHD is observed in these plasmas with frequency range of 7-20 kHz, although its impact on energy confinement or other discharge features appears insignificant. The high density flattop experiments routinely sustained H-modes into rampdown with ICRF heating, confirming dedicated rampdown studies at higher field of 5.4 T. Rampdown experiments have demonstrated that maintaining H-mode into the rampdown phase is a viable technique to avoid an over-current in the OH1 coil in C-Mod (or CS1 coils in ITER), and that the plasma current should be ramped down faster rather than slower in order to guarantee that the OH coil current is decreasing in the rampdown phase when the H-L transition does occur. The density is found to decrease with Ip in these regulated H-modes, making n/n_{Gr} constant, and both the density and temperature pedestals show a progressive decrease as Ip drops. The ultimate H-L transition in rampdown occurs at $P_{net}/P_{thr,LH}$ of 0.37-0.46 for n/n_{Gr} = 0.54-0.58 at 5.4 T, and 0.7-0.9 for n/n_{Gr} = 0.75-0.89 at 2.7 T, although the experience on C-Mod indicates that the threshold power does not scale directly with the toroidal field.

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References

[1] E. S. Marmar et al Fusion Sci Tech **51** 2007 261, entire issue.
[2] C. E. Kessel et al Nuc Fus **49** 2009 085034.
[3] S. P. Hirshman and G. H. Neilson Phys Fluids **29** (1986) 790.
[4] S. Ejima et al Nucl Fusion **22** (1982) 1313.
[5] W. A. Houlberg Nucl Fusion **27** (1987) 1009.
[6] Y. Martin et al J Physics: Conference Series **123** (2008) 012033.
[7] M. Brambilla, "A full wave code for ion cyclotron waves in toroidal plasmas",Report IPP 5/66, Max-Planck-Institut fur Plasmaphysik, Garching, 1996.
[8] Y. Ma et al Nuc Fusion **52** (2012) 023010.
[9] J. W. Hughes et al Nuc Fusion **51** (2011) 083007.
[10] X. Q. Xu et al Phys Plasmas **10** (2003) 1773.
[11] G. Saibene et al Phys Plasma Control Fusion **44** (2002) 1769.
[12] J. E. Rice et al Plasma Phys Control Fusion **50** (2008) 124042.
[13] O. Sauter et al Phys Plasmas **4** (1997) 1654.
[14] R. J. Buttery et al Nucl Fusion **43** (2003) 69.
[15] A. C. C. Sips, Nucl. Fusion **49** (2009) 085015

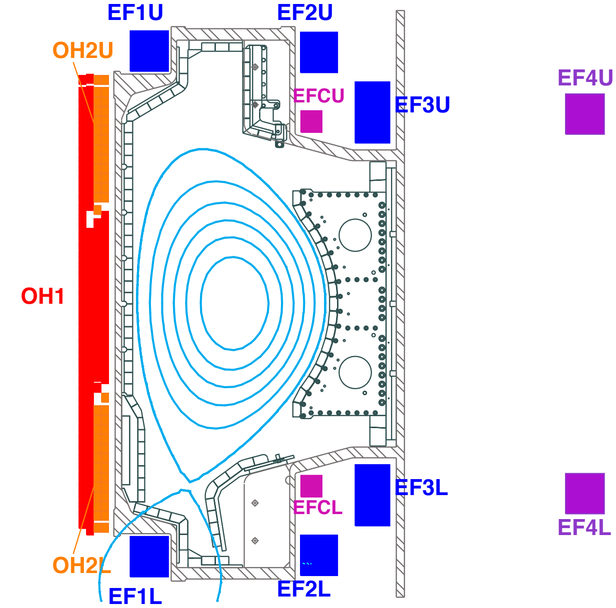

Figure 1. Cross-section of the Alcator C-Mod tokamak, displaying the OH and PF coils, dominant conducting structures, and an ITER-like plasma poloidal flux contours from a TSC simulation.

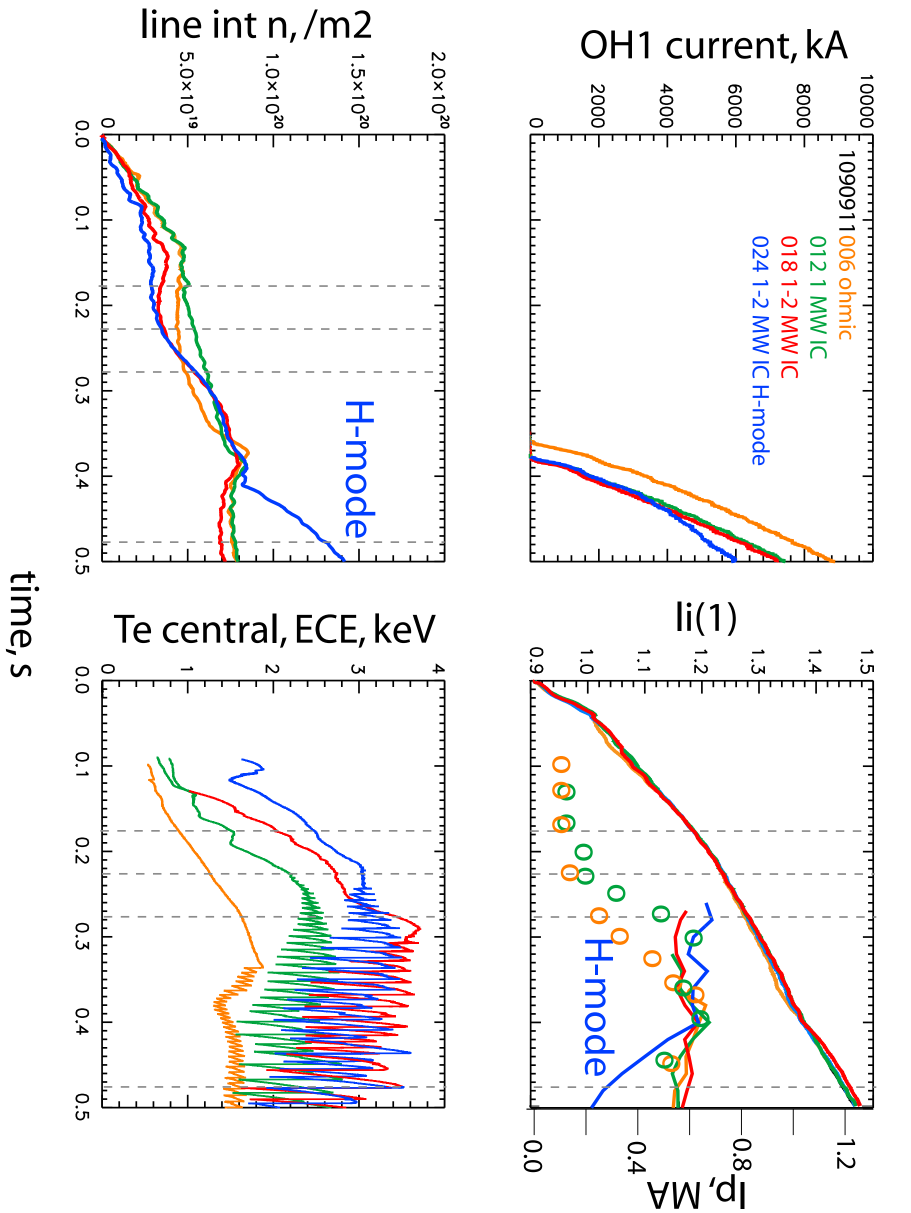

Figure 2. Time histories of the OH1 coil current, li(1) and plasma current, line density, and central temperature for four plasma current rampup experiments, showing the reduced swing of the OH1 coil current as the plasma is heated in L-mode, and with an H-mode transition. Open circles on the li plot indicate improved equilibrium reconstructions relative to the automated reconstructions shown by the solid lines. The vertical dashed lines are the time-slices for temperature profiles shown in Fig. 4.

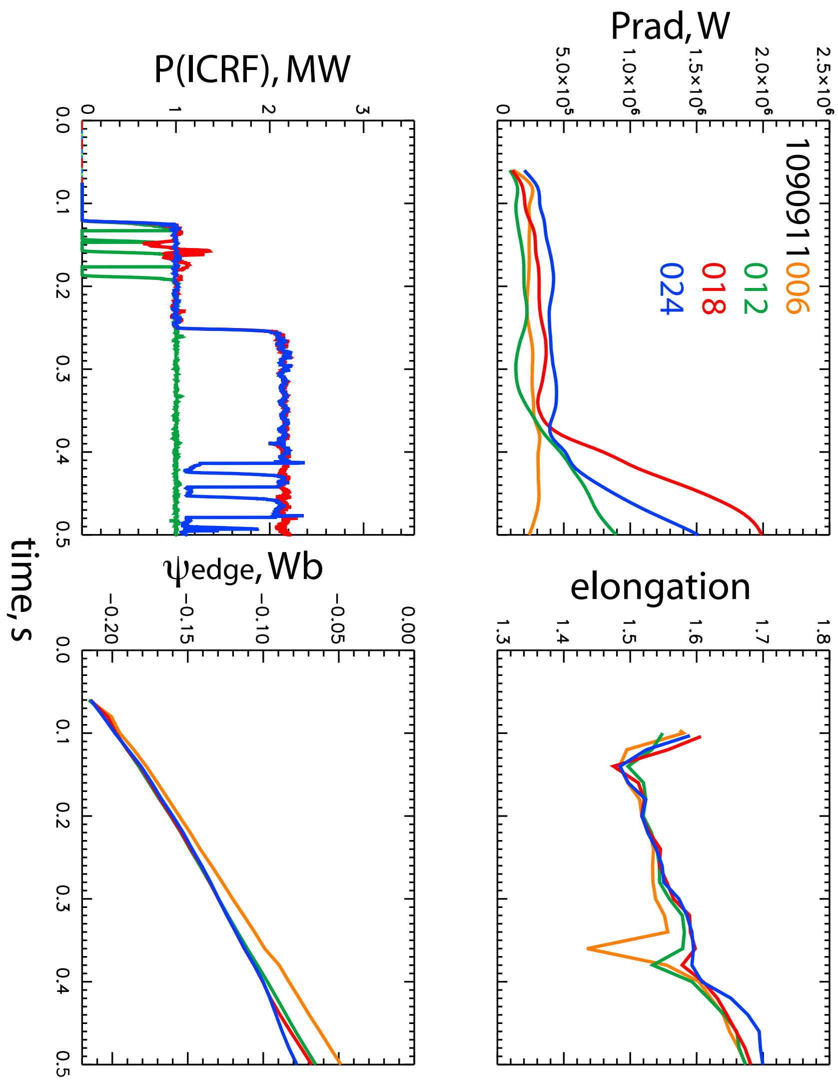

Figure 2. Time histories of the plasma radiated power, plasma elongation, ICRF injected power, and plasma edge poloidal magnetic flux for the four current rampup experiments. The strong perturbation seen in the plasma elongation plot is due to the OH coil cross-over.

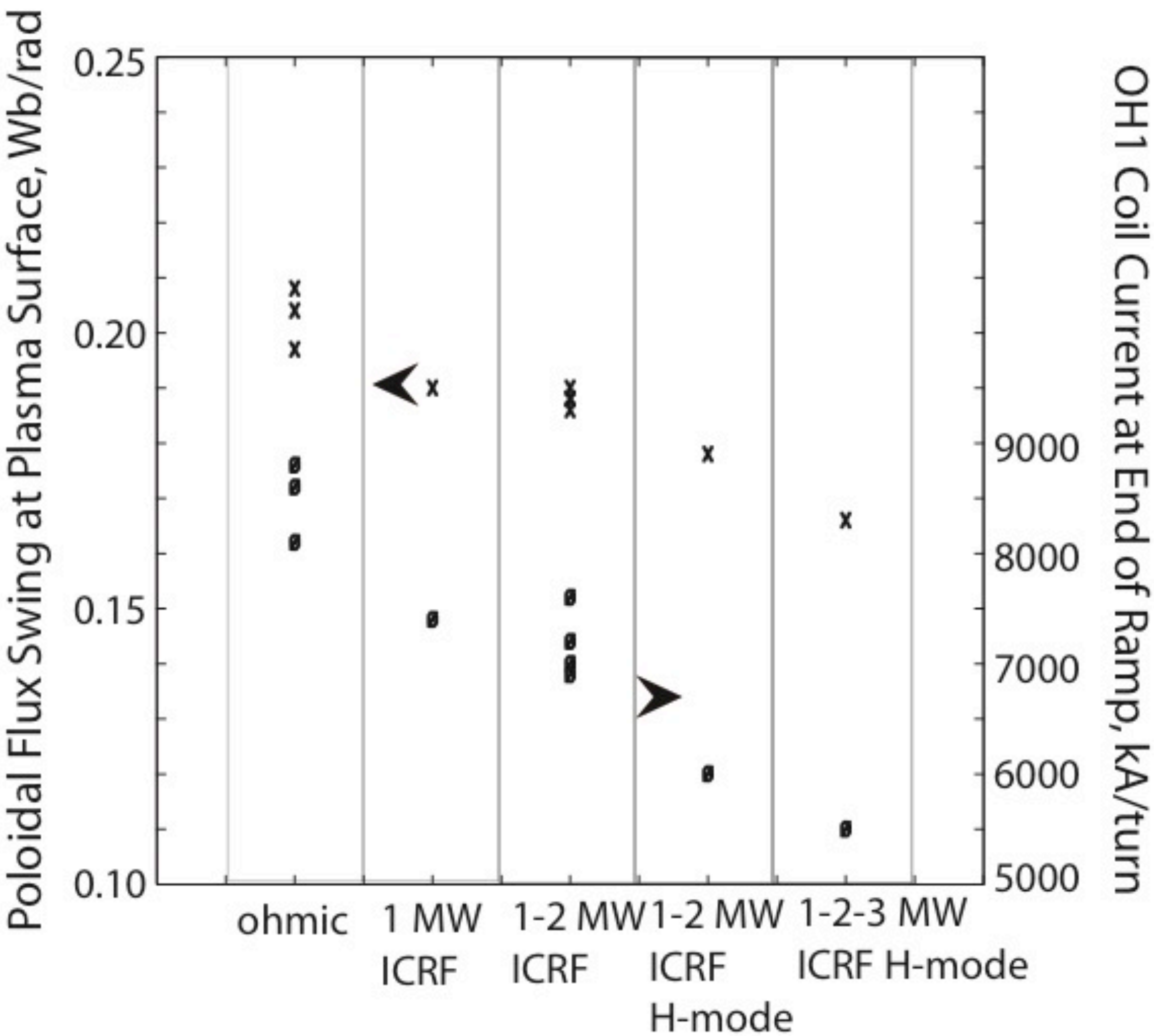

Figure 3. Plasma edge poloidal magnetic flux swing during the plasma current rampup, and OH1 coil current at the end of the current rampup, for ohmic rampup, ICRF 1 MW heating in L-mode, ICRF 1-2 MW heating in L-mode, ICRF 1-2 MW heating with an H-mode transition at 4/5 of the rampup, and a projected case with ICRF 1-2-3 MW heating and an H-mode transition at 4/5 of the rampup (which is a projection).

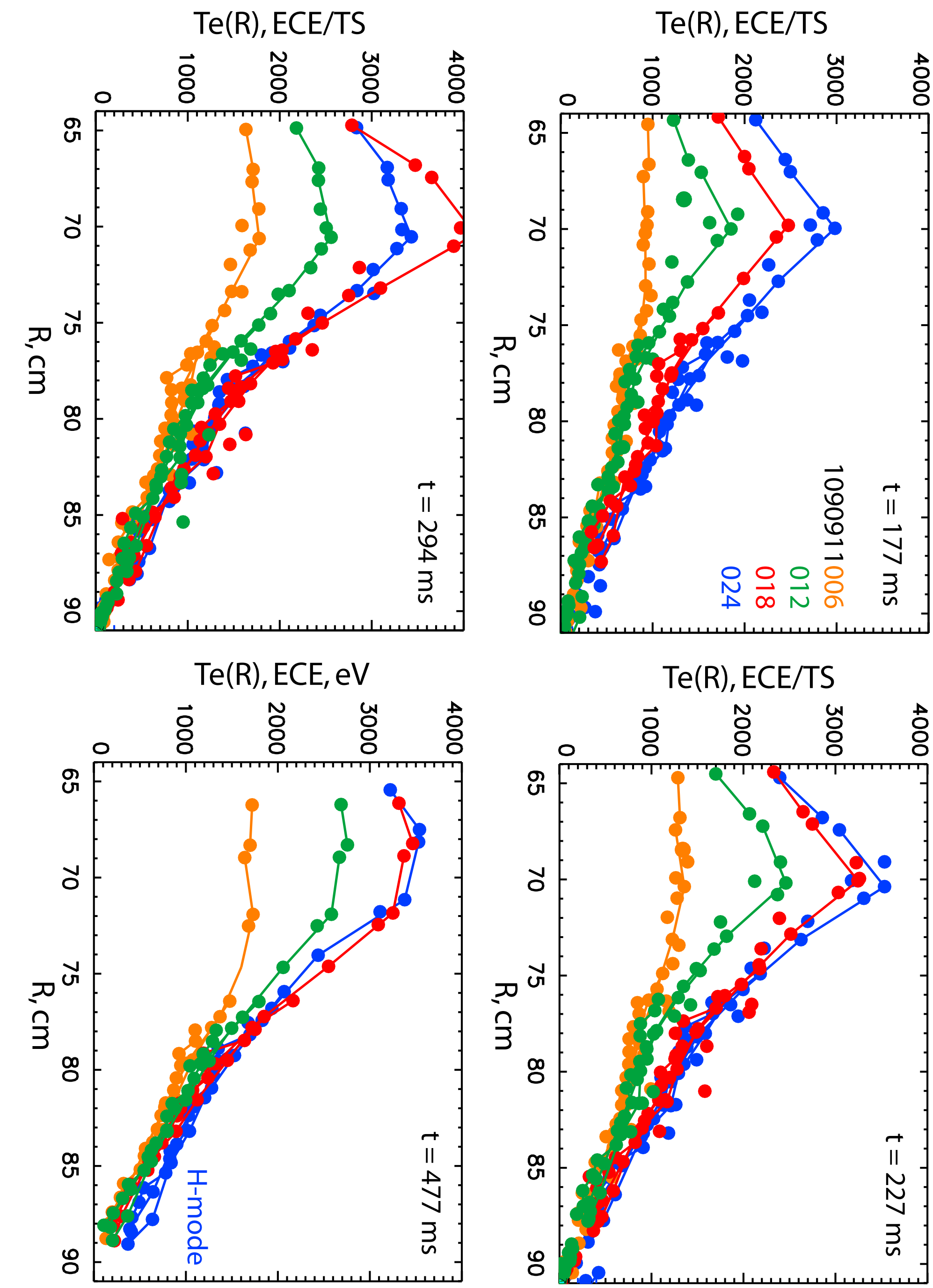

Figure 4. The electron temperature profiles from Thomson scattering and electron cyclotron emission for the ohmic (orange), 1 MW ICRF heated L-mode (green), 1-2 MW ICRF heated L-mode, and the 1-2 MW heated with H-mode transition (blue), at four time slices. These show the strong peaking in the temperature profiles with ICRF heating, which subsequently lead to earlier sawtooth onset.

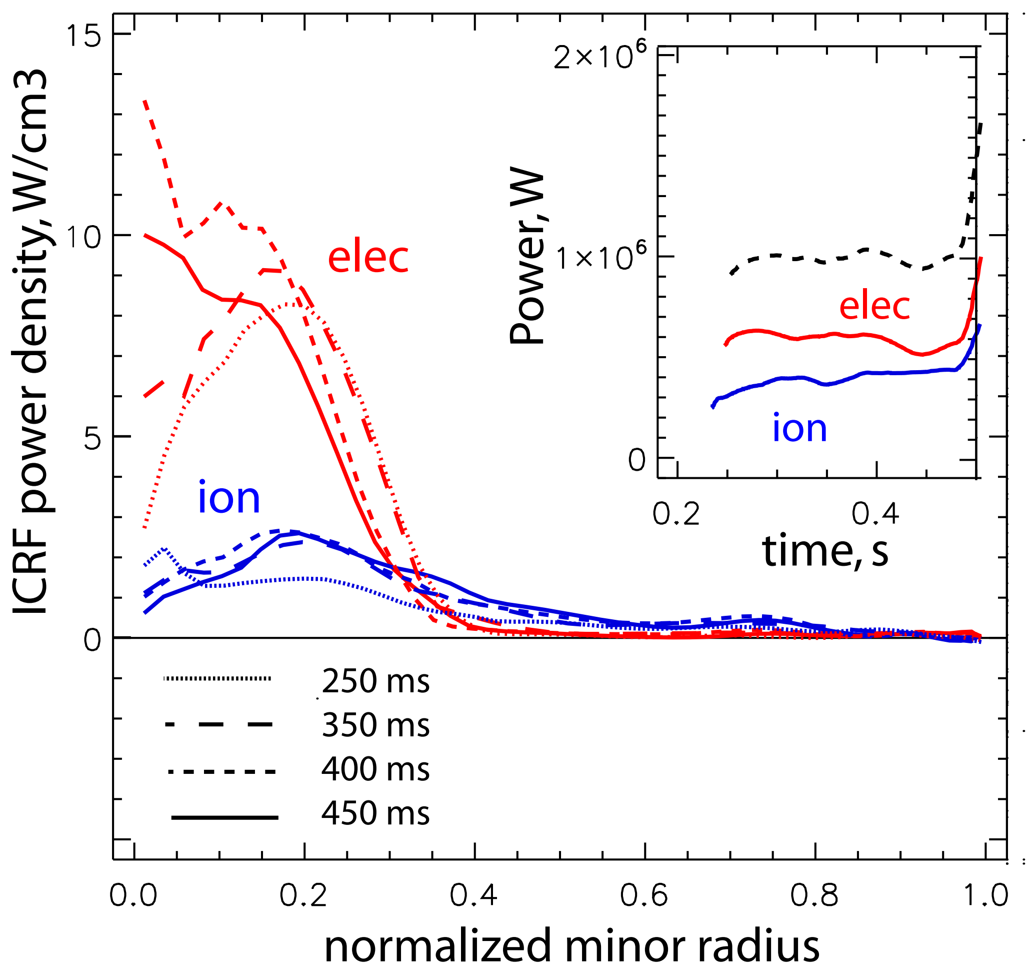

Figure 5. ICRF heating profiles as function of normalized minor radius for thermal electrons and thermal ions at four time slices in the 1 MW ICRF heating in L-mode discharge, showing the strong central deposition characteristic of the hydrogen minority (assumed 5% of total ions in analysis) heating in these experiments. Also shown are the total powers deposited on electrons and ions during the rampup phase, showing slightly higher electron power at lower density, reaching a 55/45 split at the end of the rampup.

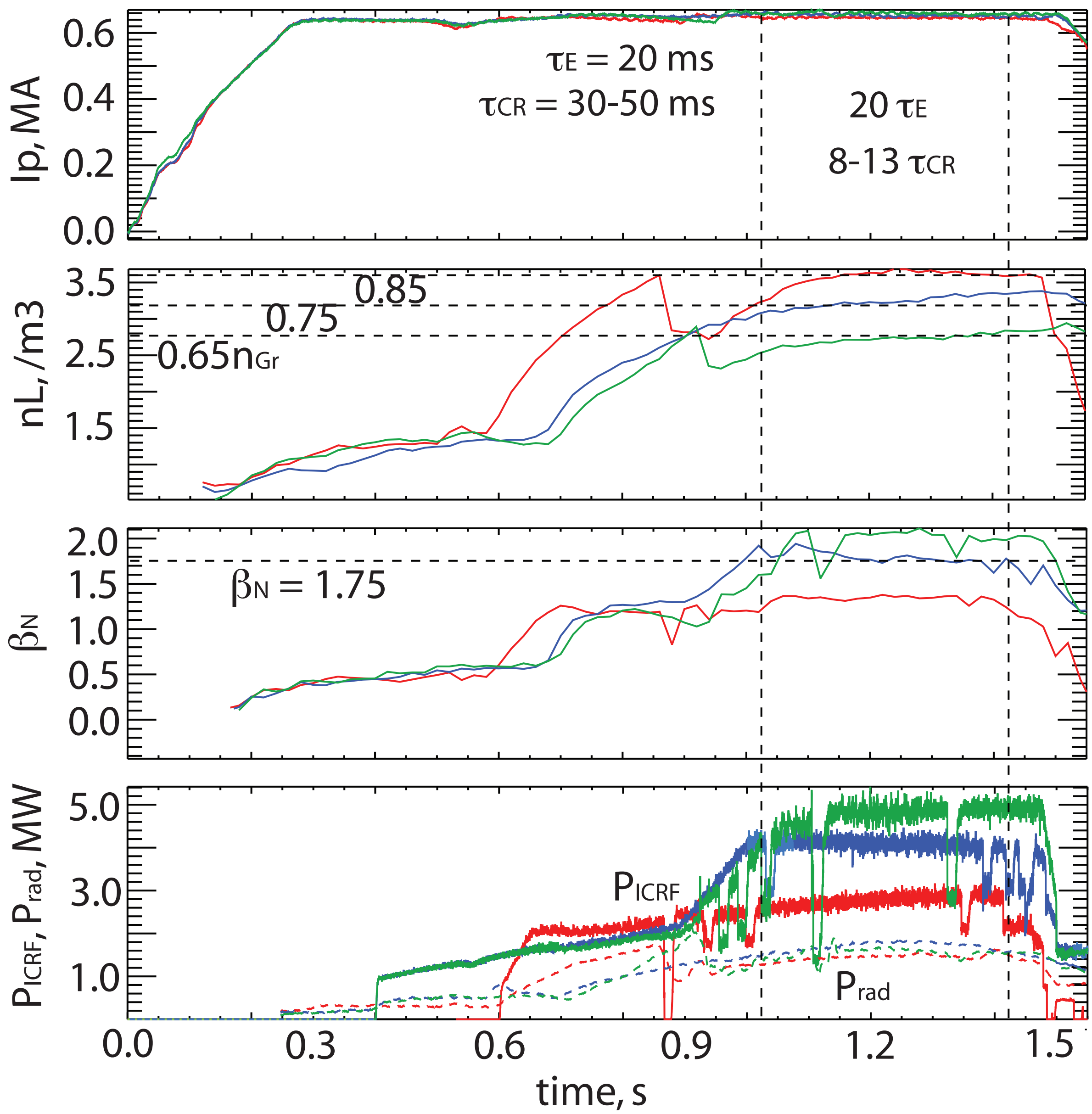

Figure 6. Time histories of the plasma current, the line average density, normalized beta, and both ICRF heating and plasma radiated powers for three discharges that represent the range of results obtained in flattop experiments at 2.7 T. The quasi-stationary phase where scalar data is obtained is shown, spanning 20 energy confinement times and ~ 10 current diffusion times.

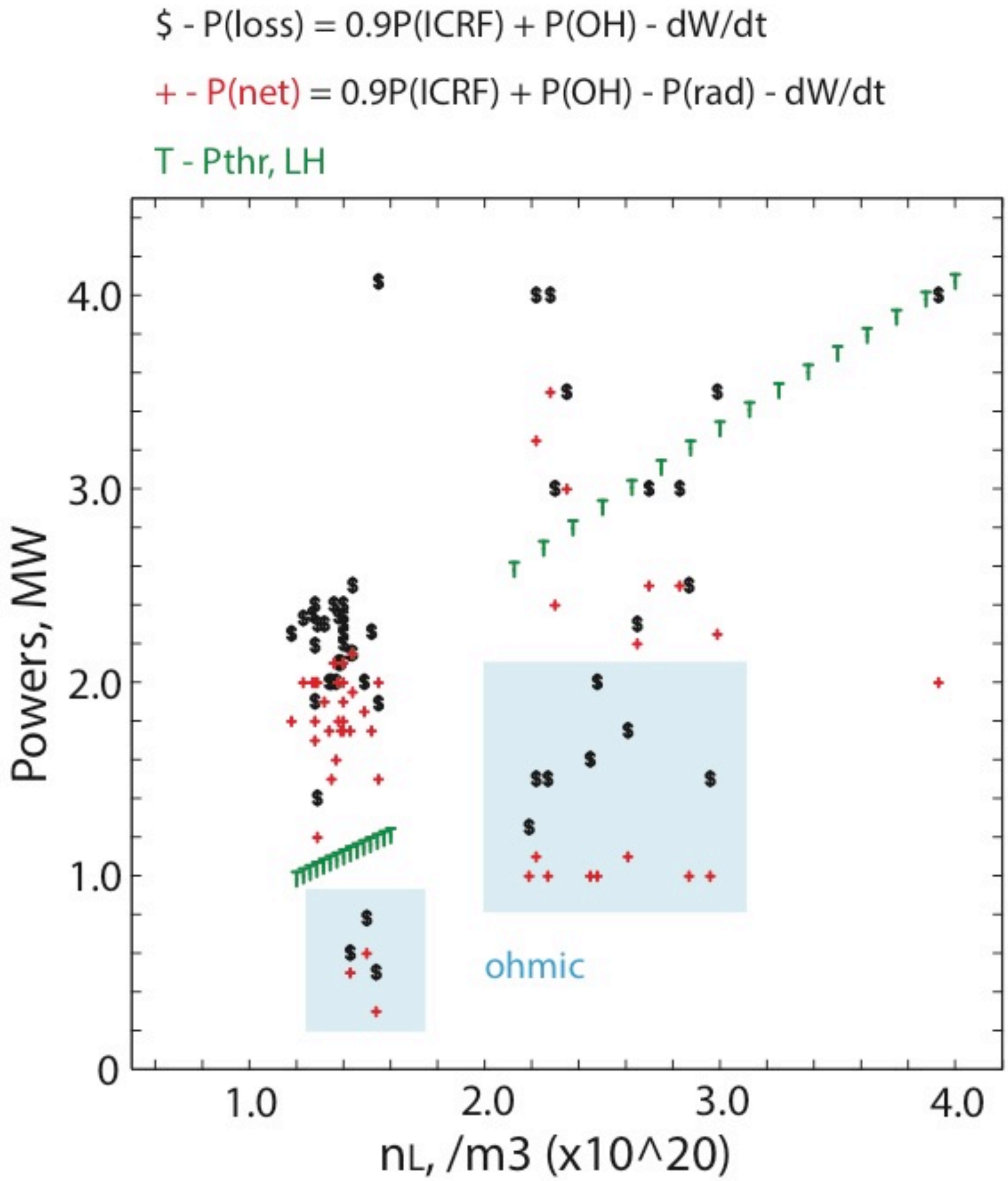

Figure 7. Onset powers for the H-mode at 2.7 T (left, lower), and 5.4 T (right), reporting the loss power (black), the net power (red) and the corresponding threshold power (green). Regions shaded in blue are ohmic only H-mode onsets.

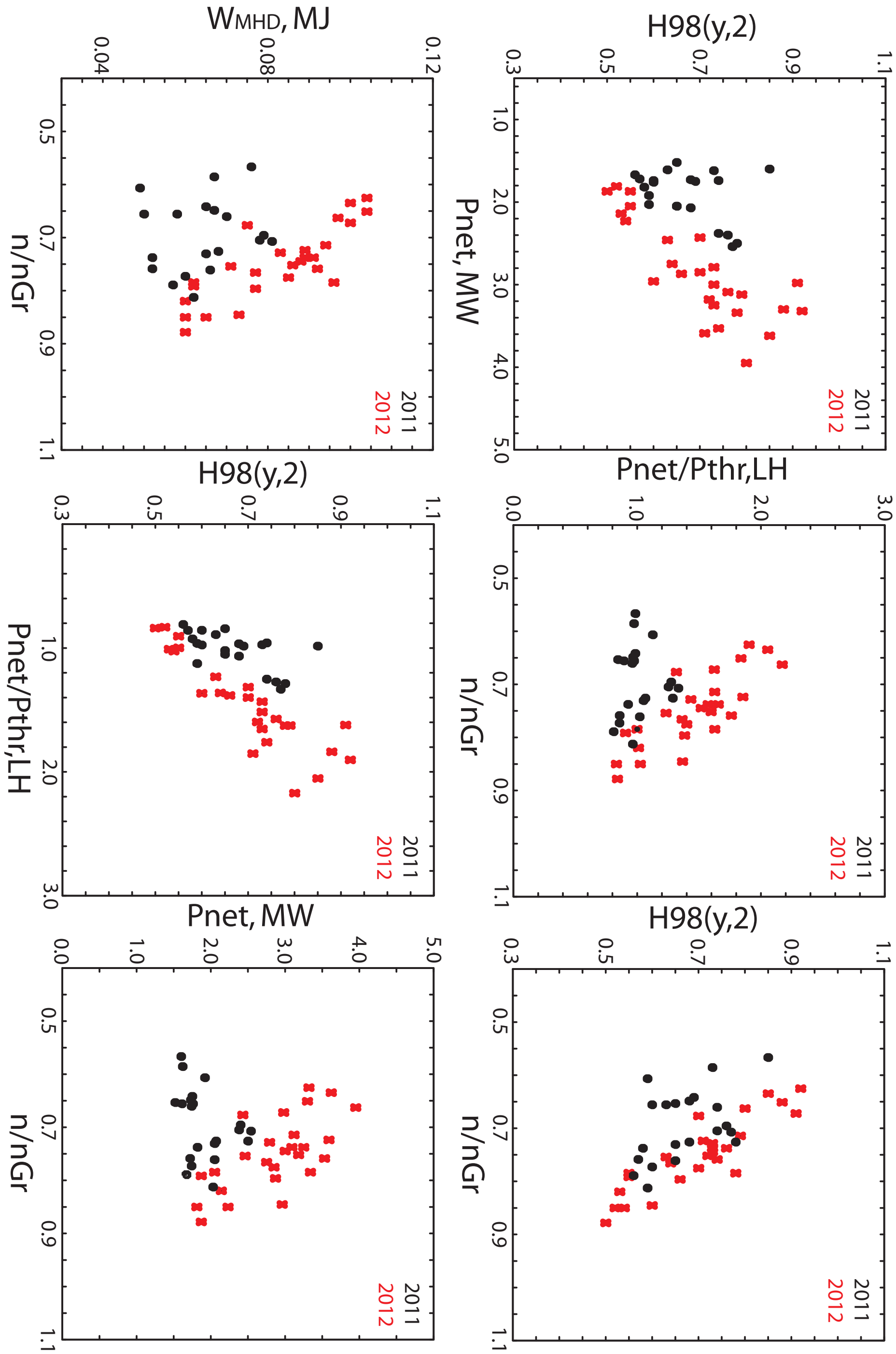

Figure 8. Scalar data taken in the quasi-stationary phase of high beta and high density discharges, both from 2011 and 2012 campaigns. The energy confinement multiplier for IPB98(y,2), net power and ratio of net power to LH threshold power, stored energy, and n/n_{Gr} are shown against each other. The sharp reduction in energy confinement with n/n_{Gr} above 0.6 is evident, as well as the reduction of net power with the highest densities.

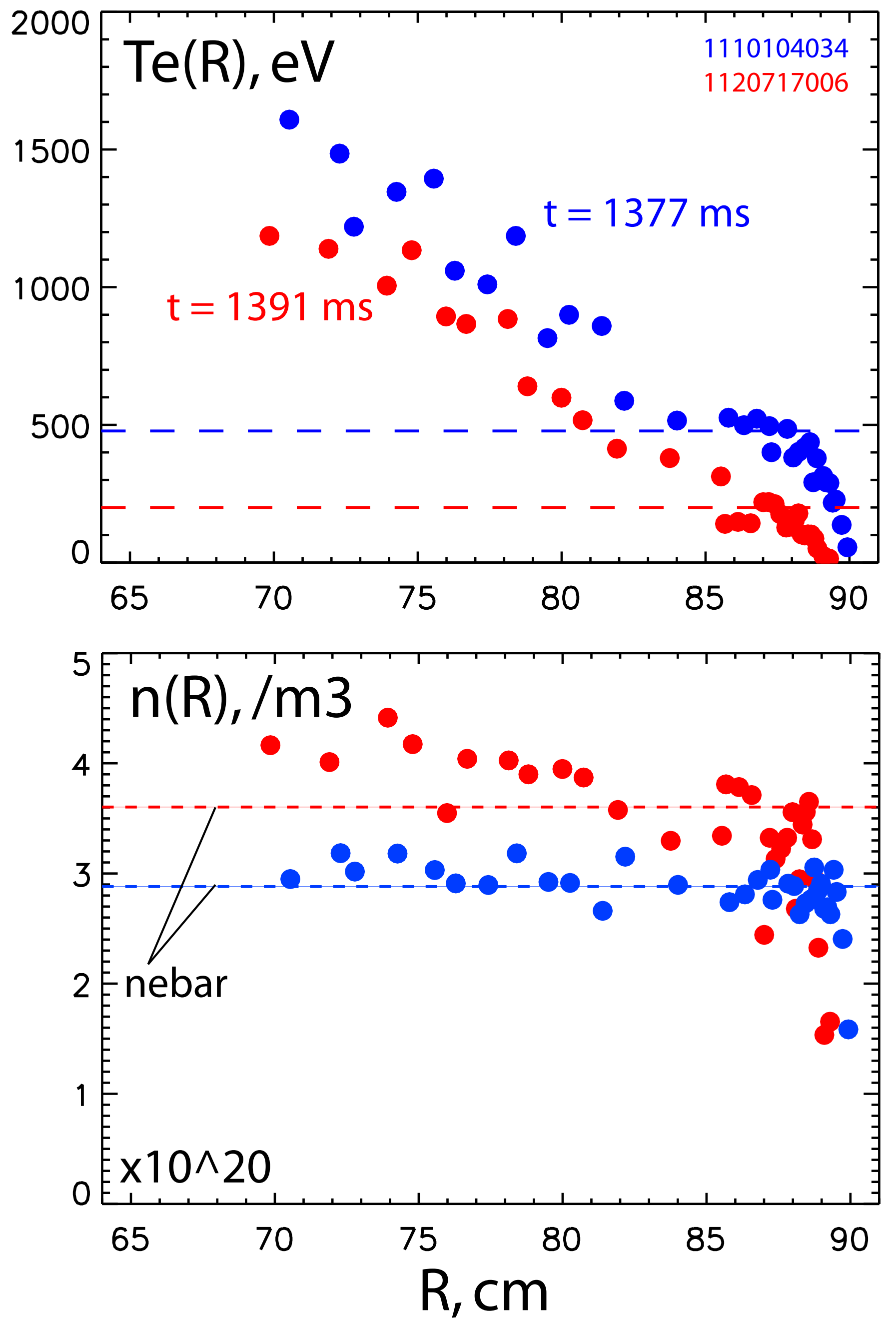

Figure 9. Temperature and density profiles for two high beta and high density discharges with nearly identical ICRF and net powers, but one with $n/n_{Gr} = 0.68$ and $H_{98(y,2)} = 0.88$, and the other with $n/n_{Gr} = 0.84$ and $H_{98(y,2)} = 0.53$. In spite of an ~ 30% increase in the density throughout the profile, the core temperature drops by $\leq$ 30%, but the pedestal drops by 2.5 times. The plasma edge is undergoing a strong cooling at this density.

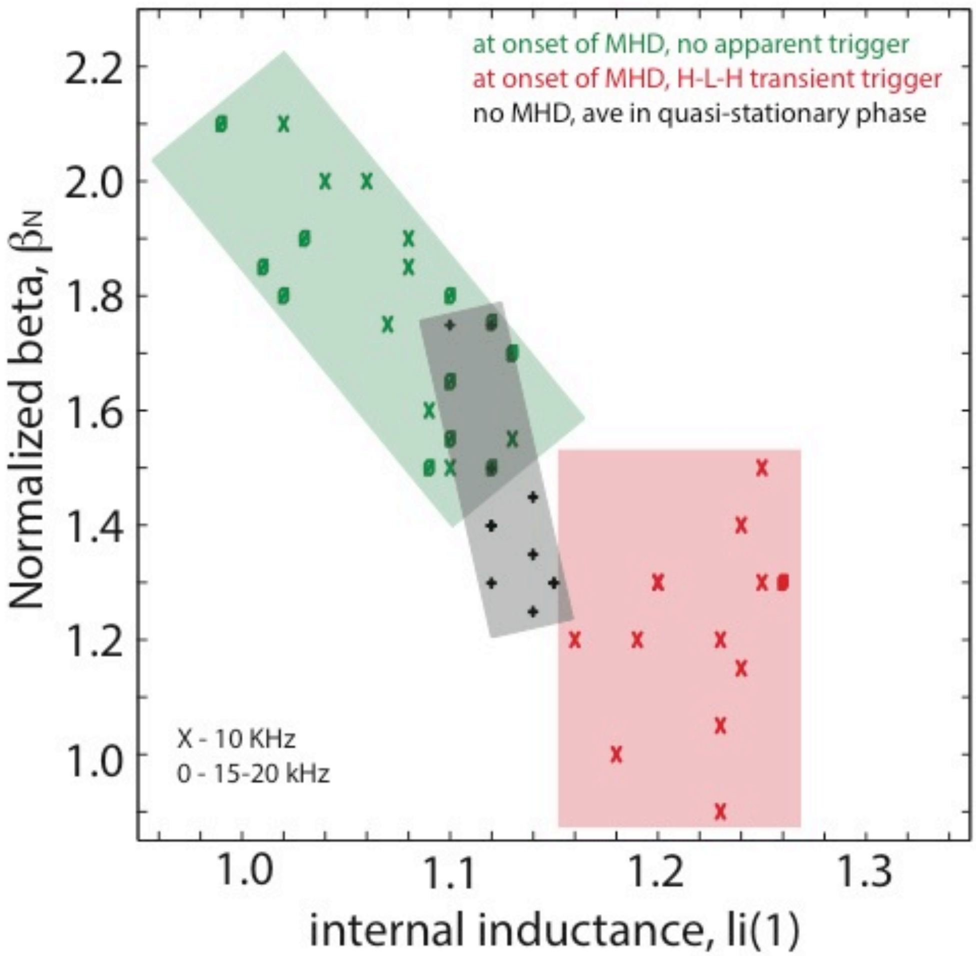

Figure 10. Normalized beta versus internal self-inductance li(1) at mode onset for transient induced modes (red) and spontaneous onset (green). The values, averaged during quasi-stationary flattop phase, are also shown for the discharges which did not have a mode (black).

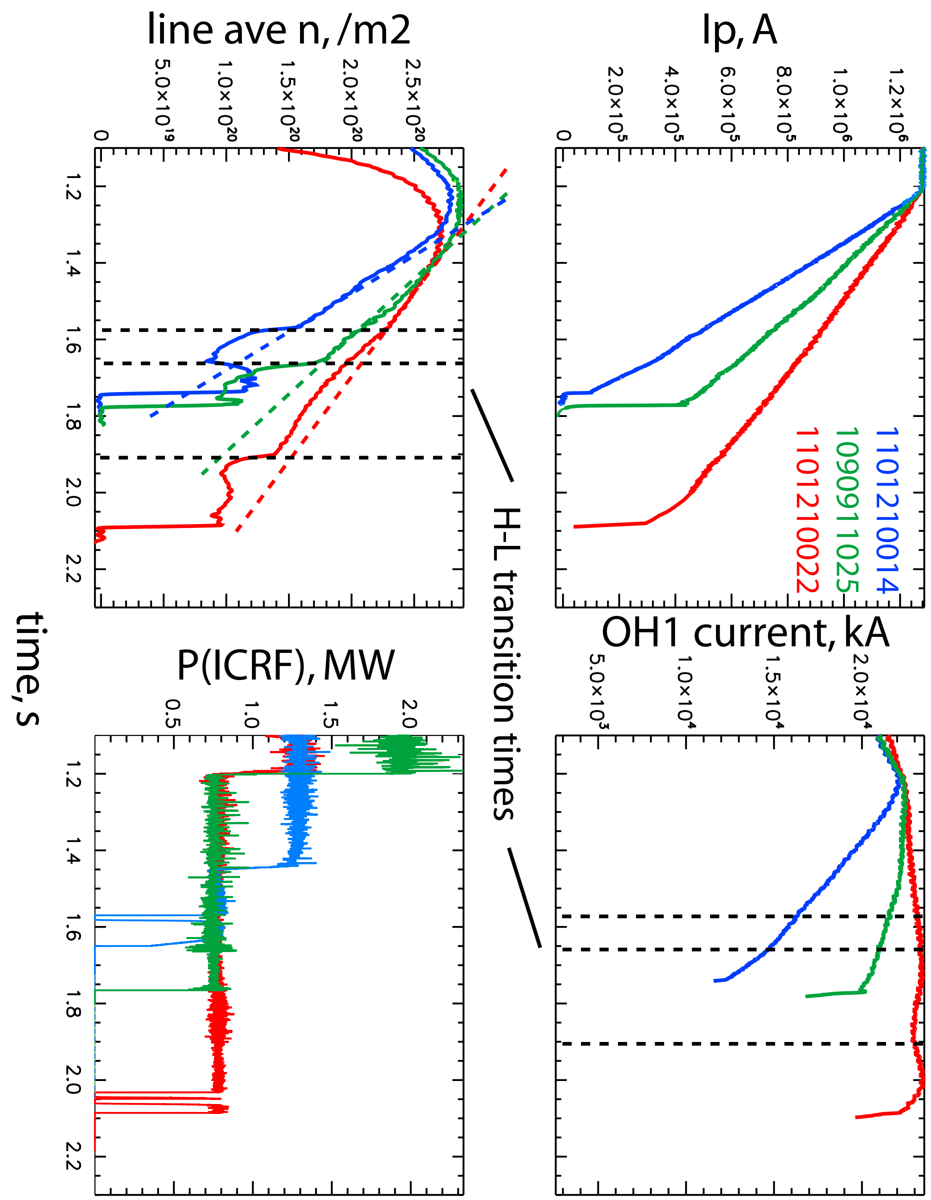

Figure 11. Time histories of the plasma current, line average density, OH1 coil current and the ICRF heating power for three rampdown discharges with varying rampdown times. All cases have sustained EDA H-modes into the rampdown, and experience an H-L back transition at least ½ way into the rampdown duration. The vertical dashed lines are the location of the H-L back transition.

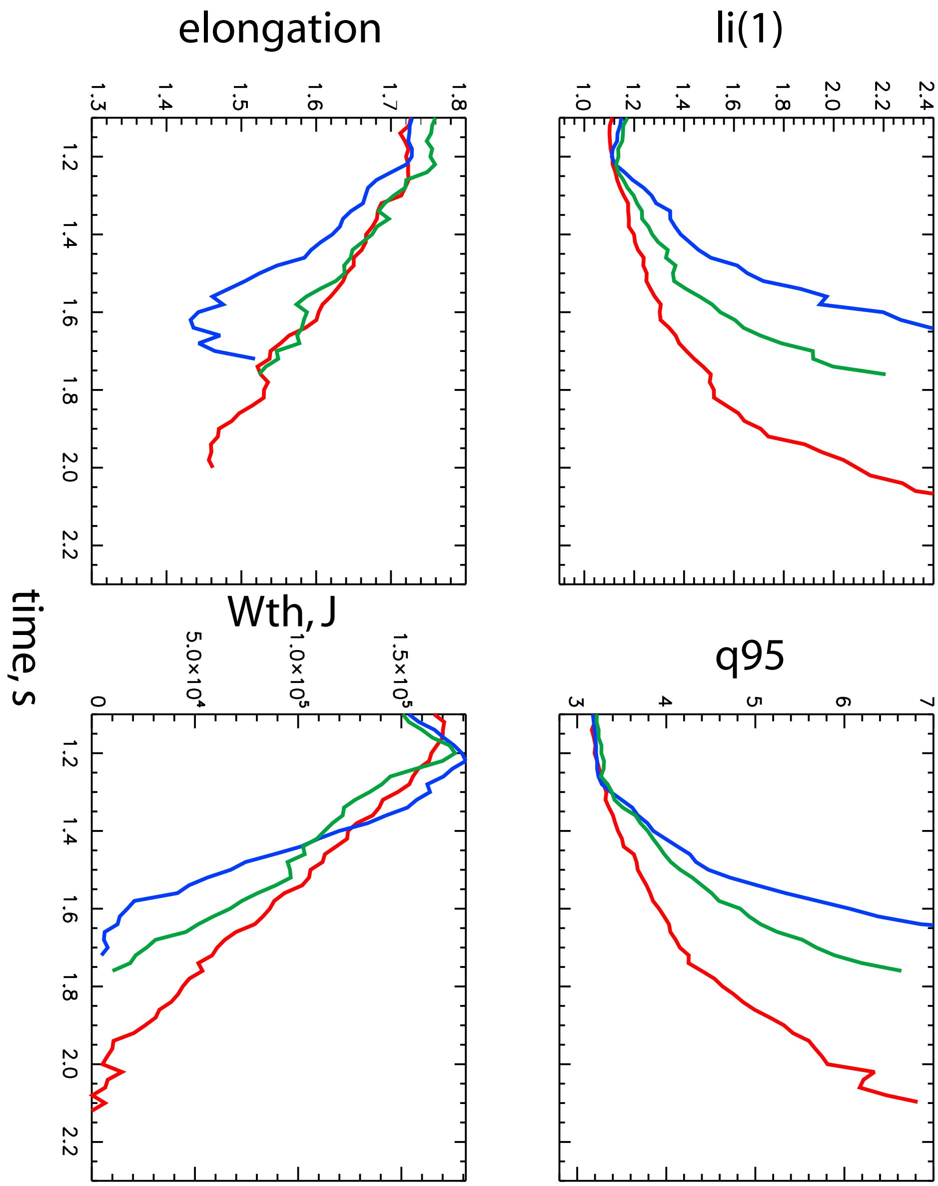

Figure 11. Time histories of the li(1), q_{95}, plasma elongation, and plasma stored energy for the rampdown discharges with 3 rampdown times.

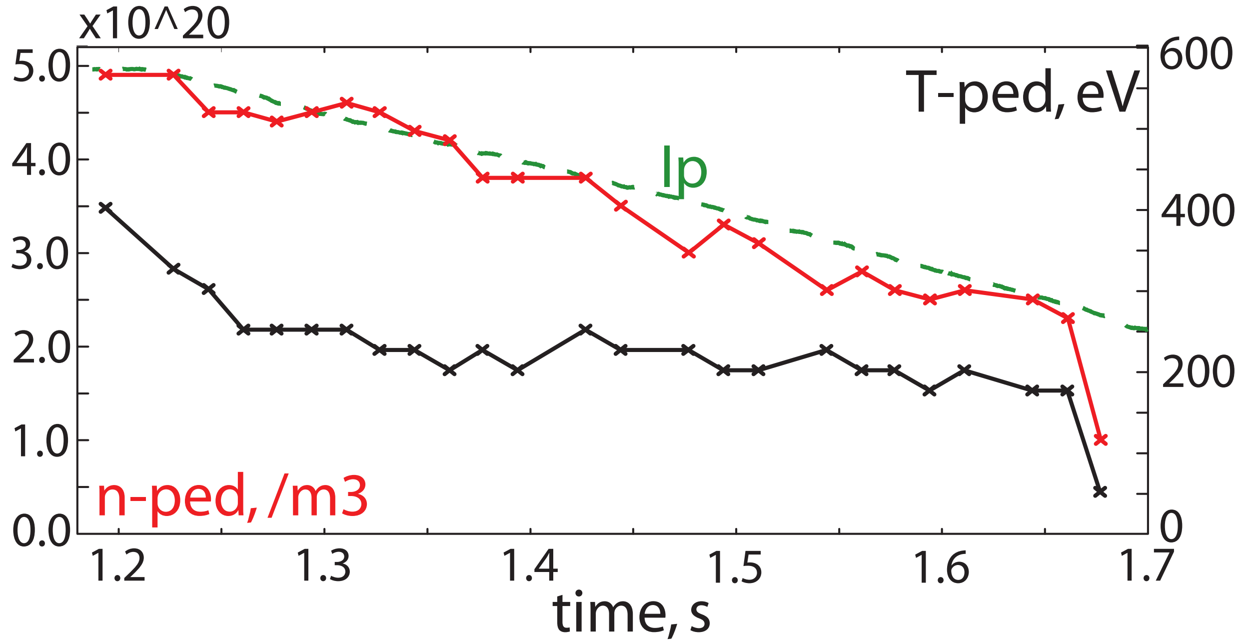

Figure 12. Time history of the pedestal temperature and density from Thomson scattering, showing the density reduction with Ip, and with a fast temperature drop early followed by a very slow reduction. The L to H transition subsequently occurs at about 1.66 s.

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