Compatibility of ITER Scenarios with full Tungsten Wall in ASDEX Upgrade

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Abstract

In 2008, ASDEX Upgrade has started its second experimental campaign with full tungsten coverage of the plasma facing components. In going from a partially W coated device (in 2004/2005) to a full tungsten device (in 2007), a reduction by an order of magnitude in both the carbon deposition and deuterium retention was found. The tungsten source is dominated by sputtering from intrinsic light impurities. The outer divertor is by far the strongest tungsten source region, but tungsten influxes from the outboard limiters most efficiently reach the plasma core. The application of ICRH results in large W influxes due to light impurities accelerated by electrical fields at the ICRH antennas. In H-mode discharges, central heating is used to increase turbulent diffusion avoiding tungsten accumulation and is provided by neutral beams and the upgraded ECRH systems. ELMs are important in reducing the inward transport of tungsten in the H-mode edge transport barrier and are controlled by gas puffing. Even without boronization, stationary, ITER baseline H-modes (H_{98}~1, β_N~1.6-2), with W concentrations below $3 \cdot 10^{-5}$ were routinely achieved up to 1.2 MA plasma current. The compatibility of high performance improved H-modes with un-boronized W wall was demonstrated, achieving H_{98}=1.1-1.2 and β_N up to 2.6 as required for advanced scenarios in ITER. Further ITER discharge scenario tests performed in 2008 include the demonstration of low voltage plasma start-up with ECRH and heating during the current rise to q_{95}=3, to achieve a range of plasma internal inductance of 0.71-0.97. The new results reported here form the basis to further enhance the operational range of ASDEX Upgrade with the full tungsten wall, support the ITER like wall project in JET as well as the ITER design, and prepare for ITER operation.

1. Introduction

Future fusion reactors will have to rely on high-Z plasma facing components to provide low erosion, a low tritium inventory and stability against neutron damage [1]. However, ITER will start with a conservative path, using a large fraction of carbon and beryllium walls and carbon target plates together with tungsten at the divertor entrance, an approach to be tested in JET by the ITER like wall project [2]. In the nuclear DT phase a transition to a full tungsten device is envisaged. The main aim of the ASDEX Upgrade tungsten programme is to qualify the use of high-Z materials for future fusion devices investigating the plasma wall interaction, the plasma operation and the compatibility with high performance scenarios and heating methods in an all-W divertor tokamak. Since 1999, the plasma facing components (PFCs) in ASDEX Upgrade have been progressively changed from carbon to tungsten coated tiles [3]. Before the 2007 campaign, the last remaining carbon elements in the main chamber and the divertor strike point tiles were exchanged [4,5]. Utilizing 200µm W-coatings on fine grain graphite substrates on areas of high erosion and 3-5 µm elsewhere, a full tungsten plasma facing interior was created. After damaging of one of three flywheel generators in April 2006, operation in 2007 was restricted to 800 kA with ≤7.5 MW of input power. For 2008, the remaining flywheel system was enhanced, allowing 1MA at ≤16 MW, providing a more stringent testing environment for the full tungsten wall compared to 2007. Diagnostic improvements in 2007-2008, support the tungsten programme and aid in a better characterisation of the plasma wall interaction and plasma performance [6-11].

This paper describes the transition to a full tungsten device, plasma operation schemes used to achieve stationary H-modes with full W wall, the optimisation of improved H-modes performance and specific ITER discharge scenario studies at ASDEX Upgrade.

2. Transition from carbon to a full tungsten device

The tritium inventory in the presently proposed ITER wall material mix is expected to be dominated by co-deposition with carbon. In ASDEX Upgrade the step by step transition to an all-W machine has been monitored by measuring the deposition of carbon in the inner divertor and remote areas. Post campaign analyses of first wall tiles show that during the whole campaign (3000s) in 2004/05 14 g of C were deposited, in 2005/06 this was reduced to 2.2 g and for the 2007 campaign the carbon deposition is down to 1g [12]. During experimental campaigns, the carbon concentration is monitored using spectroscopic measurements in similar discharges. The C concentration in the pedestal is observed to reduce from peak values of 2% (with some carbon surfaces present) to 0.5%-1% in the full tungsten machine [13]. The carbon potentially comes from the erosion of not fully coated tile gaps, small regions where the tungsten coating has been eroded away and the carbon content in the tungsten coatings. Hence, in a carbon free device, the deposition would reduce further.

The avoidance of boronisations during the 2007 and the initial 2008 campaign provided a clean test bed for the post campaign investigation of the deuterium retention [13, 14]. Accurate data are obtained using surface analyses techniques on tiles removed from the vessel. Typically 4% of the injected gas was retained for carbon PFCs, mostly found in re-deposited carbon-deuterium layers. Only 0.3% fuel retention is found for full W-coated PFCs [12]. This reduction by an order of magnitude is required to provide sufficient experimental time (>10000 discharges) in ITER before reaching the tritium inventory limit. Detailed analyses show that the D-inventory in the full-W ASDEX Upgrade is dominated by co-deposition in remaining carbon layers on the high field side (HFS) target tiles, the remaining part originates from diffusion of deuterium into the W layers of the LFS divertor [5]. The removal of the few remaining boron layers from the walls provided clean conditions for new experiments on hydrogen retention in 2008, using gas balance analysis on a shot to shot basis. These new results show that ~3.6%±3% of the deuterium gas is retained 3 minutes after the discharge. However, only ~1.5%±3% is retained during the high density phase of the discharge, a relevant number for extrapolation to long pulse operation in ITER. Clear wall saturation is observed, at very high gas input levels (above $\sim 4\cdot 10^{22}$ atoms). However, similar high gas flux data are not available for the carbon phase of ASDEX Upgrade, as discharges with less gas input were typically performed in the full W device. After the first boronization in April 2008 the D inventory increased again due to co-deposition with B, putting a question mark on the use of Be in the first ITER phase.

The line radiation has been routinely measured for several campaigns monitoring the change from a carbon to a tungsten machine. Up to 10^4 foil bolometers provide input for de-convolution methods to reconstruct the radiation in the divertor and main plasma. Before completion of the full W coverage, both the total radiated power and the distribution of the radiation emissivity re-mained more or less independent of the ratio between tungsten and carbon coated PFCs [11]. Since 2007, a significant reduction of the radiation below/near the X-point region is found due to the decrease of the carbon content. This is

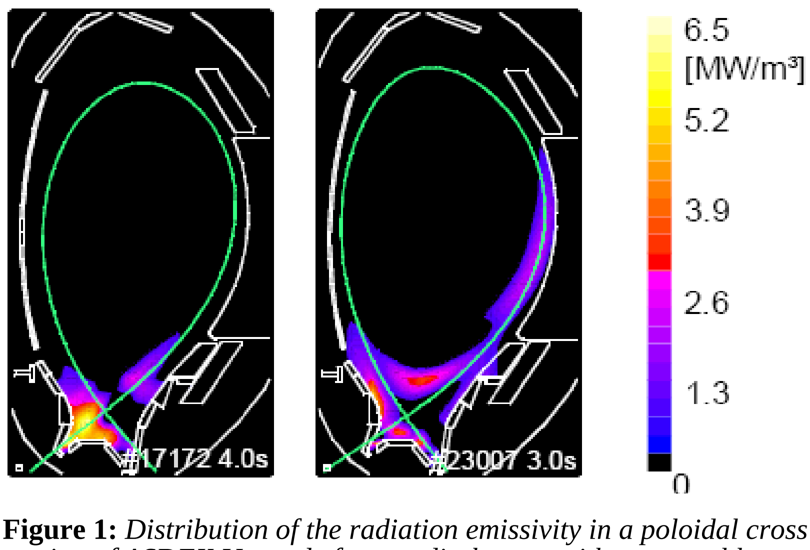


Figure 1: *Distribution of the radiation emissivity in a poloidal cross section of ASDEX Upgrade for two discharges with comparable plasma parameters. On the left, with mostly carbon PFCs. On the right, with full W covered PFCs.*

demonstrated in Fig. 1 showing a comparison of ELM averaged radiation distributions for two similar 1 MA discharges - an older discharge with nearly all carbon PFCs and a recent discharge with tungsten coated PFCs. In contrast the radiation from the main plasma and the edge has increased in the unboronized W device mainly caused by ~1% oxygen content. About 60% of the input power is found as radiated power, of which only 5-10% is located in the divertor and 45% in the main plasma.

3. Plasma operation

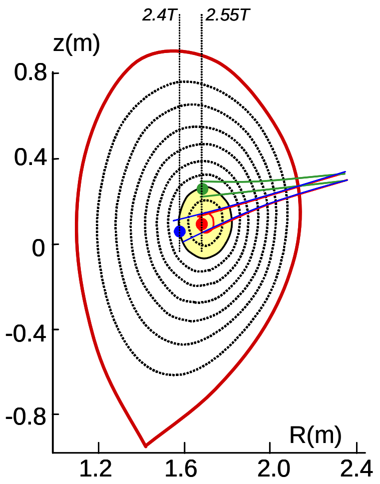
In 2007, three plasma restarts after vacuum breaks were successfully performed without boronisations, to demonstrate the behaviour of a "true" high-Z device in preparation for long pulse operation in ITER and DEMO. Also the 2008 campaign was restarted without a boronisation, applying the same conditioning procedures as used in 2007 [4], i.e. a long (203 hour) bake-out of the vessel at 150°C and several overnight helium glow discharges. Further conditioning was performed with ~50 plasma discharges using moderate heating power of up to 7.5 MW, together with inter-shot helium glow discharges. These conditioning discharges have an elevated helium content. A strong degradation of the confinement is found for helium concentrations in the main chamber of about 5 - 10%, with H_{98}~0.6-0.8 [15]. The storage and release of helium after He glow discharges is significantly different for W plasma facing components compared to graphite, as confirmed in laboratory experiments [16]. Hence, the inter-shot glow discharges in helium were replaced by occasional short (2 minute) glows in deuterium. Using this technique, the helium concentration of the plasma was reduced to <1% within 10 discharges, restoring H_{98}~1 in stationary H-mode discharges of 6 s. This milestone was set after 17 operation days including the implementation of an optimized use of the generator power after damage of one flywheel generator.

To assess tungsten erosion, the thickness of the W coating is measured after the campaign using Rutherford-backscattering [17], giving a campaign integrated value for the erosion. For the 2007 campaign the average erosion rate at outer divertor location was determined to be 0.12 nm/s for a total of 2317 s of divertor operation. Time resolved tungsten erosion is measured using spectroscopic measurements of the W I emission in the divertor and main plasma chamber with a time resolution of 0.2-3 ms, sufficient to measure erosion during and in between ELMs [5]. The largest erosion rates of tungsten due to plasma impact are observed at the strike point tile of the outboard divertor [5]. In discharges with high divertor temperature (>12 eV, as measured by Langmuir probes), the tungsten influx (Γ_W) in between ELMs can be as high as $1 \cdot 10^{20}$ m^{-2}s^{-1} near the outer strike put. This is ~50% of the Γ_W averaged over the ELM cycle. For discharges at high recycling, with colder divertor temperature (<6 eV), $\Gamma_W \sim 0.1 \cdot 10^{20}$ m^{-2}s^{-1} is measured in between ELMs. This is 10%-20% of the averaged tungsten influxes near the outer divertor during the ELM cycle. Consequently, the contribution of the ELMs in the erosion mainly depends on the divertor temperature in between ELMs. The measured sputtering yields increase sharply above 12 eV temperature. A response not expected in pure deuterium plasma, but rather from conditions with a few % of low-Z impurities. Running the machine unboronised, the main chamber radiation from intrinsic impurities has been sufficient to keep the temperature in the divertor low enough (<20 eV). Hence, to date nearly all discharges have omitted impurity seeding. Exploratory experiments with Argon seeding show that the W source in between ELMs is reduced due to the lower divertor temperatures obtained (~5 eV) [6]. However, during ELMs the tungsten erosion is slightly enhanced, due to sputtering by seed ions. The inner heat shield, on the high field side, is usually the first limiting structure of the scrape-off layer (SOL) plasma, and consequently produces larger W influxes compared to the outboard limiters. Nonetheless, influxes from the outboard limiters are the most important source for the tungsten content in the plasma. When using ICRF heating schemes the active antennas produce a strong local increase (by an order of magnitude) of the tungsten influx [5,18].

4. Heating schemes and stationary H-mode operation

In order to obtain stationary H-mode operation, the tungsten concentration needs to be controlled in the plasma. In H-mode, the edge transport barrier provides a zone of good confinement, and the inward transport of tungsten is regulated by the ELMs. A higher ELM frequency is beneficial for reducing the tungsten concentration. The ELM frequency is determined by the amount of additional heating and gas puff level used. The optimisation of the plasma performance implies a control of the tungsten concentration while maintaining good plasma confinement (see section 5). In the core of the plasma, the neo-classical inward flow of W can lead to accumulation [6] of the tungsten favoured by density peaking at high confinement conditions. Central heating needs to be provided by neutral beam injection and RF heating to provide enough central heat flux to enhance the turbulent transport in the centre of the plasma. Careful combination of these tools allows stable operation with tolerable core W concentration c_W below $5 \cdot 10^{-5}$.

The ECRH system on ASDEX Upgrade is currently being extended by 4 MW for 10 sec, using 4 gyrotrons with frequencies at 105 GHz and 140 GHz [19]. One new, long pulse (10 s) gyrotron was available for the recent campaigns, together with 4 older units at 500kW/2s each. With full W-coverage, the ECRH system with its localized deposition is crucial in suppressing central accumulation of tungsten in H-mode discharges [20]. Especially in discharges were (without ECRH) the central heat flux provided by neutral beam injection is not enough to overcome the neo-classical inward flow of W. However, application of 140 GHz ECRH, using second harmonic X-mode is restricted to central densities $\leq 1.2 \cdot 10^{20}$ m^{-3}, only compatible with operation at $\langle n_e \rangle / n_{GW} \leq 0.80$ for 1MA discharges at ASDEX Upgrade. The narrow and flexible power deposition allowed dedicated experiments to investigate the influence of the central heating [21]. The ECRH resonance position was changed from ρ_{tor}=0.04 to $\rho_{tor} \geq 0.2$ by steering the launch angle poloidally or by changing the toroidal field away from 2.55 T (Fig. 2a). For $\rho_{tor} \geq 0.2$, a rapid (within 0.2-0.5 s, a few energy confinement times) accumulation of central impurities can be observed. Fig. 2b shows that the tungsten accumulation occurs inside ρ_{tor}=0.3, with power densities reaching 1-2 MWm^{-3}. Hence, the requirement is to position the ECRH resonances close to the axis. The reason for the extreme sensitivity to the resonance location is still somewhat unclear. In addition to enhancing turbulent transport by increasing heat flow, (1,1)-MHD activity may be involved in suppressing impurity accumulation. This (1,1)-MHD activity is typically located near ρ_{tor}=0.2 in these H-mode discharges.



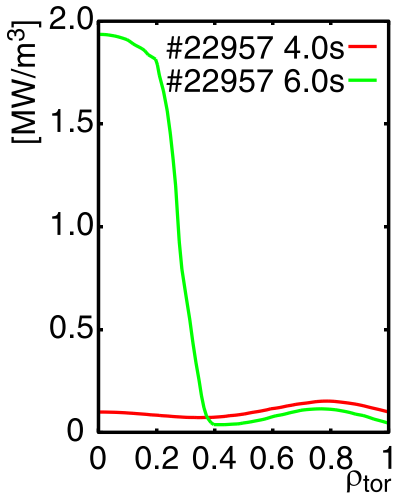


Figure 2a: *Scan of the ECRH deposition radius in the centre of the plasma. Using central deposition (red injection geometry) the tungsten accumulation in the core can be controlled. Steering the ECRH deposition vertically up (green injection geometry), or moving the deposition inboard by changing the toroidal field (blue injection geometry) leads to uncontrolled rise of the tungsten content in the centre.*
Figure 2b: *Radiation profile in the main plasma as a function of the normalized flux radius during phases without ECRH heating (green) and with ECRH heating (red). The central radiation was kept successfully low by applying ECRH, inside* ρ_{tor}*=0.2.*

ICRF operation with high-Z plasma facing components leads to enhanced local erosion by up to an order of magnitude of PFCs connected along magnetic field lines to the active ICRF antennas [18]. The cause lies with the high RF electric fields parallel to magnetic field lines, and the corresponding sheath rectified (DC) RF potentials on the field lines. These voltages mainly originate from the antenna near-fields, rectified by the plasma, accelerating hydrogen ions as well as light impurity ions, such as oxygen, leading to the observed enhanced sputtering of tungsten. Apart from boronisation, the tungsten source from ICRF can be reduced by creating low temperature conditions at the plasma facing components using a large clearance between the separatrix and the antenna (>6 cm) and by elevated gas puff rates ($\Phi_{D,puff}$> 5•10^{21}/s). Operation of neighbouring ICRF antennas at the phase difference close to -90 degrees can also lead to a reduction in W source [18]. Nevertheless, a reduction of near-fields by antenna design is needed to minimize the tungsten source to acceptable levels. Code calculations predict a dominant role of currents within the box surrounding the antenna in the formation of antenna near-fields. Corresponding evidence has been found in experiments, providing a good basis for designing an improved antenna for ASDEX Upgrade.

The optimisation of stationary H-modes, is performed by adjusting the level of gas fuelling and ECRH power at a given neutral beam power and q_{95}. Above minimal requirements (for ECRH~0.5 MW, for $\Phi_{D,puff}$~4•10^{21}/s) the amount of ECRH and fuelling are exchangeable; i.e. for higher ECRH power (>1 MW), the gas fuelling can be reduced, while for higher gas fuelling, $\Phi_{D,puff}$>10•10^{21}/s, the ECRH power can be reduced. In H-modes, clear trends are observed: (1) At lower q_{95}, more gas fuelling or central ECRH is needed, (2) at higher total input power from NBI heating, less gas fuelling or central ECRH is needed, (3) the ELMs are essential in reducing the inward transport of tungsten in the H-mode edge transport barrier. In most discharges, the ELM frequency is accelerated by applying a higher gas puff, which reduces energy and impurity confinement at the same time. In the future, ELM triggering by small pellets will be used to allow independent control of the ELM frequency, recycling and the radiation by seeded impurities. Despite the restriction set by the available generator power, discharges at 1.2 MA/2.55 T were carried out in the unboronised W vessel, allowing the use of ECRH at q_{95}<4 [21]. These ITER relevant H-modes (see Fig. 3) show H_{98}~1, β_N~1.6-2.0, <n_e>/n_{GW}~0.65 and with a radiation fraction of about 60% (without impurity seeding), W concentrations below 3.10^{-5} and a carbon content of about 0.5%, providing confidence that ITER can reach its goal of Q=10, even with a full tungsten wall.

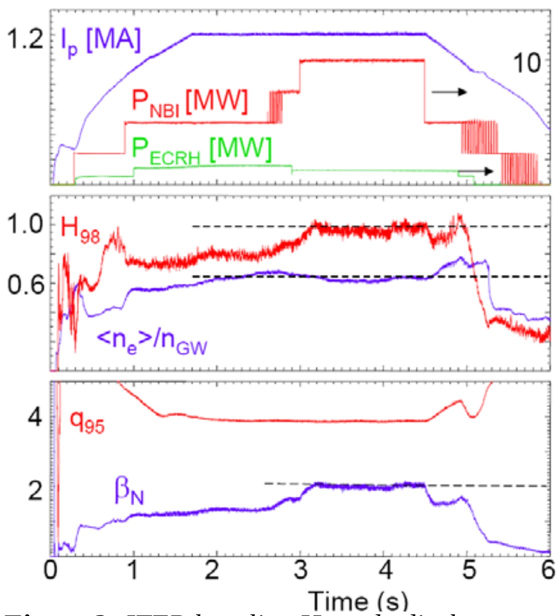


Figure 3: *ITER baseline H-mode discharge compatible with un-boronized full W wall at 1.2 MA (q95=3.8) with H_{98}~1, β_N~2, <n_e>/n_{GW}~0.65 and with a radiation fraction of about 60%.*

5. Compatibility of high performance improved H-modes

"Improved H-mode" discharges in ASDEX Upgrade are characterized by confinement factors H_{98}>1, total beta β_N = 2-3.5 and a q-profile with ~zero shear in the core of the plasma with q(0)~1 [22]. Before 2007, the highest confinement factors (H_{98}=1.2-1.4) were obtained at the lowest values of the plasma collisionality achievable at low or zero gaspuff level [23]. In 2008, experiments have concentrated on establishing stationary high performance H-mode discharges without boronisation in a full W wall. The optimisation of the plasma performance included a selection of the type of neutral beam sources used in the experiments (on-axis or off-axis). Although off-axis NBI provides better control of low shear q-profile [23], central neutral beam injection was used to maximise the power deposition in the core to assist in the prevention of tungsten accumulation. Typically, ECRH is applied for 4 s at ~1.6 MW, deposited within ρ_{tor}=0.2 and gas fuelling from the main chamber is used. In otherwise constant conditions, this gas fuelling rate was varied from a high level, $\phi_{D,puff} = 15 \bullet 10^{21}$/s, to a low level of $\phi_{D,puff} \leq 2.0 \bullet 10^{21}$/s in successive discharges. Fig. 4a shows the variation of the stored energy for this deuterium fuelling scan, at 10 MW-14 MW NBI power and 1.6 MW ECRH. The results are compared to improved H-modes obtained in previous campaigns without deuterium fuelling. The stored energy increases when the deuterium fuelling is reduced, reaching ~1 MJ at $\phi_{D,puff} = 3.0 \bullet 10^{21}$/s. This is the lowest deuterium fuelling rate for which stable tungsten concentrations during the discharge are obtained, using the available 1.6 MW ECRH. The plasma density increases with increasing deuterium gas fuelling, hence the density in the 2008 discharges is substantially higher, $\langle n_e \rangle / n_{GW}$ ~0.65, compared to previous campaigns. Fig. 4b shows the variation of the confinement enhancement factor with the plasma density, normalised to the Greenwald density. A reduction of H_{98} at higher densities (higher deuterium fuelling rates) is seen. The variation of the confinement enhancement factor with density is more marked compared to the variation in stored energy (Fig 4a), as the confinement scaling predicts poorer confinement at lower density [21]. At the lowest deuterium fuelling rates (and lowest plasma density or collisionality) H_{98}~1.1-1.2 is achieved, in line with results from previous campaigns. Higher confinement or plasma energies might be achievable even with high densities at higher triangular plasma shapes as shown in [24].

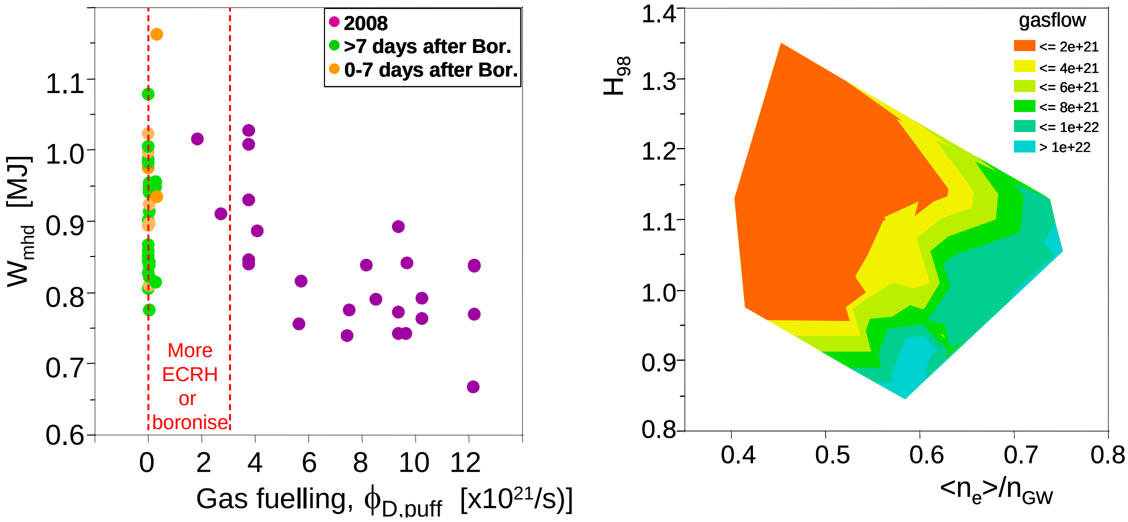


Figure 4: *On the left, the stored energy dependence on the level of gas fuelling used. Discharges from the 2008 campaign are with gas fuelling, while all discharges from 2003-2006 have zero gas fuelling levels. Discharges done within 7 calendar days of a boronisation (orange circles) and afterwards (green circles) are shown. On the right, the confinement factor achieved versus line-averaged density normalised to the Greenwald density. The contours indicate the level of gas fuelling (1/s) used in these discharges. All discharges in the two overview plots are at 1MA, 2.3-2.6T, δ<0.32 and total additional heating power 10-14 MW.*

During the 2008 campaign, routine pedestal profiles have been obtained [8]. An Integrated Data Analysis method [10] combines edge lithium beam emission spectroscopy and core DCN interferometry data with a time resolution of 50 µs and a spatial resolution of ~5 mm at the plasma edge rising to ~10 cm in the core. A new CXRS diagnostic viewing the edge region delivers detailed information of the impurity density, ion temperature and toroidal rotation [9] with 1.9 ms time resolution. Higher edge densities are observed for the recent discharges with deuterium fuelling and significantly lower edge temperatures. To document the role of the H-mode pedestal on global confinement in hybrid discharges, dedicated power (total beta) scans have been performed [25]. In these dedicated experiments the H_{98}

confinement factor increases with total β_N in ASDEX Upgrade, as observed from global data base analyses of improved H-mode data [23]. The beta of the pedestal (β_N^{PED}) increases with the total beta thermal ($\beta_{N,th}$, without fast particle content). In improved H-modes the increase in global confinement is driven by the increase of the pedestal confinement for stiff plasma profiles.

In the second half of the 2008 campaign, ASDEX Upgrade has been boronised again, after a two year break from using this conditioning technique. Improved H-mode discharges have been made after boronisation. Preliminary results indicate that differences in performance (stored energy, H_{98} and beta) are <10%. With boronisation stable discharges without deuterium fuelling (achieving lower density and collisionality) are obtained, as the tungsten influxes from the outer limiter are (temporarily) suppressed. After boronizations the oxygen content is reduced again and main chamber radiation is reduced [5]. This leads to increased heat loads on the W PFCs and correspondingly higher W influx. A detailed analysis of these new results is ongoing to provide a comparison to un-boronised H-mode discharges in 2008.

6. ITER breakdown and ramp-up scenario studies

Experiments were performed on an ITER like breakdown and subsequent current rise to q_{95}=3 [26], validating these scenarios for a full tungsten first wall. For these experiments, operation without using switching resistors in the ohmic heating circuits was newly developed, reducing the loop electrical field on axis down to E~0.25 V/m. (ITER plans to use 0.32 V/m). ECRH was used for pre-ionisation at the 2nd harmonic X-mode (105 GHz at 1.7 T, or 140 GHz at 2.3T) and fundamental O-mode (105 GHz at 3.2 T), positioning the resonance on the high field side. ECRH alone (≤1 MW) or a combination of ECRH and ICRH (up to 400 kW coupled power using fundamental hydrogen resonance heating) has been used in the pre-ionisation phase. Successful pre-ionisation at power levels up to 1 MW (mainly ECRH) was achieved, without damage of the tungsten surfaces, or diagnostics by ECR stray. The fundamental O-mode experiments at 105 GHz (3.2 T) achieved the best pre-ionisation and subsequent current rise phase, and are analogous to using the main 170 GHz gyrotrons for breakdown assist in ITER at 5.2-5.3 T. Using low voltage schemes, breakdown was still achievable after high current (q_{95}=3) disruptions. Good disruption recovery is generally observed in ASDEX Upgrade after the transition to full tungsten. Following the breakdown phase, continued ramping to full current (1.0 MA at 1.7 T) in 1.0 s-1.2 s was achieved. In some experiments, ECRH was used to pre-ionise and heat the current rise at low plasma density (<3•10^{19} m^{-3}) with good plasma performance even with E<0.3 V/m (Fig. 5) and without damaging the W surfaces. The focus of the current rise experiments was on control of the plasma inductance l_i, with the various available heating schemes. NBI was used with on-axis and off axis injection up to 5 MW and ECRH was used at 0.5 MW. A clear result from these experiments is that the amount of electron heating during the current rise determines the current diffusion, with the capability of varying the internal inductance l_i significantly from 0.71to 0.97 at fixed plasma current rise rate dI_p/dt=0.66MA/s. This is within the foreseen operational boundaries (l_i=0.7-1.0) for 15MA operation in ITER given mainly from vertical position control limits.

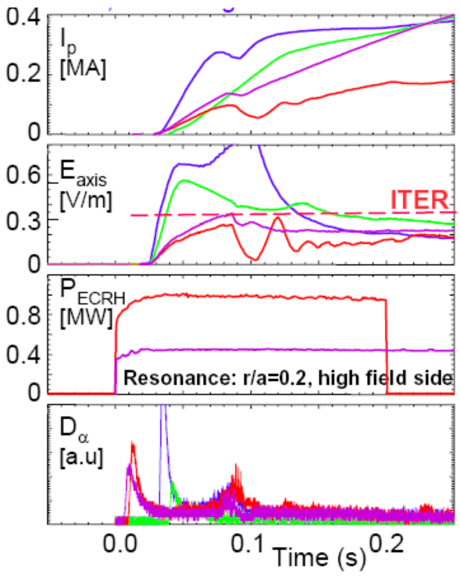


Figure 5: ECRH start up assist at AUG. The minimum required electrical field on axis is reduced to ~0.2V/m using 0.3-0.9 MW ECRH (X2).

7. Conclusions

In transition to a full tungsten device, ASDEX Upgrade has seen an order of magnitude reduction in the deposition of carbon in the divertor and remote areas, and a similar drop in the hydrogen retention. These new results are in-line with predictions for ITER, indicating that only a full tungsten wall would provide an environment with low enough tritium retention. Operating in H-mode, the outer divertor is by far the strongest source region for tungsten, especially in discharges with high divertor temperature (>12 eV) between ELMs. Nevertheless, influxes from the outboard limiters are the most important source for the tungsten content in the plasma. Hence, when using ICRH, enhanced W influxes due to sputtering from light impurities accelerated by electrical fields at the ICRH antennas, need to be reduced. For obtaining stationary H-modes, ELMs are essential in reducing the inward transport of tungsten in the H-mode edge transport barrier, the ELM frequency being controlled by gas fuelling and total input power. Other ELM control techniques are being developed at ASDEX Upgrade (pellets, internal control coils as proposed for ITER [27]) to provide better control, and to widen the operational regime of the stationary H-modes (e.g. lower density). The ability to heat in the very centre (ρ_{tor}=0.2) inside of the plasma is important to control the accumulation of tungsten. The ECRH system at ASDEX Upgrade provides this flexibility, but was limited in power to ~1.6MW for long pulse operation (4 seconds). Increasing, the ECRH power in 2009 to 4 MW / 10 s is high priority, to provide data whether the α-power in ITER would be sufficient to suppress the tungsten accumulation. Standard H-modes at ITER relevant performance (H_{98}~1, β_N~1.6-2) have been demonstrated, keeping W concentrations below 3.10^{-5}. Moreover, the achievement of improved H-modes with H_{98}>1 and β_N>2.5 gives added confidence that a tungsten wall is compatible with baseline and advanced operation in future fusion reactors. Finally, ECRF assisted ITER breakdown and current rise scenario at toroidal electric fields below 0.3 V/m were successfully tested with a full tungsten first wall.

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