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. The transition from a graphite device to a full tungsten device is demonstrated with a reduction by an order of magnitude in both the carbon deposition and deuterium retention. The tungsten source is dominated by sputtering from intrinsic light impurities, and the tungsten influxes from the outboard limiters are the main source for the plasma. In H-mode discharges, central heating is used to increase turbulent outward transport avoiding tungsten accumulation and is provided by neutral beams and the upgraded ECRH systems. ICRH can only be used after boronization as its application otherwise results in large W influxes due to light impurities accelerated by electrical fields at the ICRH antennas. ELMs are important in reducing the inward transport of tungsten in the H-mode edge barrier and are controlled by gas puffing. Even without boronization, stationary, ITER baseline H-modes ($H_{98} \sim 1$, $\beta_N \sim 2$), with W concentrations below $3 \times 10^{-5}$ 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 $H_{98}=1.1$ and $\beta_N$ up to 2.6 at modest triangularities $\delta \leq 0.3$ as required for advanced scenarios in ITER. With boronization the light impurities and the radiated power fraction especially in the divertor were reduced and the divertor plasma was actively cooled by $N_2$ seeding which is controlled by thermoelectric divertor currents. $N_2$ seeding does not only protect the divertor tiles, but also considerably improves the performance of improved H-mode discharges. The energy confinement increased to $H$-factors of 1.25 ($\beta_N \sim 2.7$) and exceeded thereby the best values, which have been obtained in a carbon-dominated machine. Preliminary investigations show that this improvement is due to higher temperatures rather than to peaking of the electron density profile. Further ITER discharge scenario tests include the demonstration of ECRF assisted low voltage plasma start-up and current rise to $q_{95}=3$ at toroidal electric fields below 0.3 V/m, to achieve a ITER compatible range of plasma internal inductance of 0.71-0.97. The new results reported here form the basis to further enhance the operational space of ASDEX Upgrade with the full tungsten wall and strongly support tungsten as a first wall material solution.

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 $200\mu m$ W-coatings on fine grain graphite substrates on areas of high erosion and $3-5\mu m$ 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 $\leq 7.5$ MW of input power. For 2008, the remaining flywheel system was enhanced, allowing 1.2 MA at $\leq 14$ MW, and 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 plasma operation schemes used to achieve stationary H-modes with full W wall and the performance optimisation of improved H-mode, the ITER hybrid scenario, both with unboronized and boronized W wall at ASDEX Upgrade. Finally specific ITER start and ramp-up discharge scenario studies are reported.

2. Transition from carbon to a full tungsten device and plasma operation

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 resulted in a strong reduction of the deposited carbon in the inner divertor and remote areas from 14 g in the whole campaign (3000s) in 2004/05 to 1 g for the 2007 campaign [12]. Post campaign investigation of the deuterium retention [13, 14] showed that typically 4% of the injected gas was retained for carbon PFCs, mostly found in re-deposited carbon-deuterium layers, while only 0.3% fuel retention is found for full W-coated PFCs [12]. These results are supported by hydrogen retention measurements from gas balance analysis showing that 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 [5]. 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 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 line radiation has been routinely measured for several campaigns with up to 100 foil bolometers monitoring the change from a carbon to a tungsten machine. Before completion of the full W coverage, both the total radiated power and the distribution of the radiation emissivity remained 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 in the unboronized W device. This is demonstrated in Fig. 1 showing a comparison of ELM averaged radiation distributions for similar 1 MA discharges. 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.

![Figure 1: Distribution of the radiation emissivity in a poloidal cross section of ASDEX Upgrade discharges with comparable plasma parameters. From left to right, with mostly carbon PFCs, then with full W covered PFCs in unboronized, boronized and boronized with nitrogen seeding](image-url)

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 plasma discharges using moderate heating power of up to 7.5 MW, together with inter-shot helium glow discharges. Those were replaced by occasional short (2 minute) glows in deuterium resulting in low helium concentrations of <1% and restoring $H_{98}$ 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.

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]. W erosion is predominantly caused by sputtering from intrinsic light impurity ions and the largest erosion rates are observed at the strike point tile of the outboard divertor during ELMs [6]. 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) even without impurity seeding. 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 [6,18].

3. 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 $5\times10^{-7}$.

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 $\leq 1.2\times10^{20}$ m$^{-3}$, only compatible with operation at $n_e/n_{GW} \leq 0.80$ 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 $\rho_{tor}=0.04$ to $\rho_{tor}=0.2$ 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 $\rho_{tor}=0.3$, with radiation 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 $\rho_{tor}=0.2$ 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 \( \rho_{\text{tor}} = 0.2 \).

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 (\( \Phi_{D,puff} > 5 \times 10^{21}/s \)). 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 \( \Phi_{D,puff} > 4 \times 10^{21}/s \)) 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, \( \Phi_{D,puff} > 10 \times 10^{21}/s \), 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 \( H_{98} \approx 1, \beta_N \approx 1.6-2.0, <n_e>/n_{GW} \approx 0.65 \) and with a radiation fraction of about 60% (without impurity seeding), W concentrations below \( 3 \times 10^{-5} \) 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.

4. Compatibility of high performance improved H-modes

“Improved H-mode” discharges in ASDEX Upgrade are characterized by enhanced confinement factors \( H_{98} > 1 \), total beta \( \beta_N = 2-3.5 \) and a q-profile with \( \sim \)zero shear in the core of the plasma at \( q(0) \approx 1 \) [22]. This scenario, which is nowadays also named as Hybrid scenario, opens the way in ITER either to much longer plasma pulse lengths with typical H-mode confinement or to improved performance with \( Q > 10 \). 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 gas puff level and after wall conditioning [23]. A major goal of the 2008 campaign was to demonstrate the compatibility of the improved H-mode scenario with a tungsten wall. First the 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. For the same reason, ECRH is applied for 4 s at \( \sim 1.6 \) MW, deposited within \( \rho_{tor} = 0.2 \), and gas fuelling from the main chamber is used to control the ELM frequency. In otherwise constant conditions, this gas fuelling rate was varied from a high level, \( \phi_{D,puff} = 15 \times 10^{21} / s \), to a low level of \( \phi_{D,puff} \sim 2 \times 10^{21} / s \) in successive discharges resulting in a decrease of the ELM frequency from \( \sim 100 \) to \( \sim 60 \) Hz. \( 2 \times 10^{21} / s \) is the lowest deuterium fuelling rate for which stable tungsten concentrations during the discharge are obtained, using the available 1.6 MW ECRH. Fig. 4a shows the variation of the stored energy for this deuterium fuelling scan, at 10 MW NBI power and 1.6 MW ECRH (1 MA/2.5T). 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 \( \sim 0.9 \) MJ. The plasma density decreases with decreasing deuterium gas fuelling, hence the density in the 2008 discharges is substantially higher, \( <n_e>/n_{GW} \sim 0.65 \), compared to previous campaigns. Fig. 4b shows the variation of the confinement enhancement factor with the gas fuelling. 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) \( H_{98} \approx 1.1 \) is achieved and the stored energy of previous campaigns could only be reached with an increased heating power of \( \sim 14 \) MW (see Fig. 4a). During the 2008 campaign, routine pedestal profiles have been obtained [8]. Edge lithium beam emission spectroscopy and core DCN interferometry data are combined with a time resolution of 50 \( \mu s \) and a spatial resolution of \( \sim 5 \) mm at the plasma edge rising to \( \sim 10 \) cm in the core [10]. 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 (β_{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.

Figure 4: Stored plasma energy and H-mode confinement enhancement H_{98(y,2)} dependence on the level of gas fuelling used for improved H-mode discharges. 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; discharges afterwards: green circles). For 2008 discharges the violet symbols refer to unboronized W walls, while the red points are with N seeding and boronized walls. All discharges in the two overview plots are at 1MA, 2.3-2.6T, δ<0.32 and total additional heating power 10-12 MW including 1.5 MW central ECRH. Confinement factors at higher triangularity δ~0.35 without boronization are also shown.

End of April 2008, ASDEX Upgrade has been boronised again, after a two year break from using this conditioning technique, which led like expected to a clear reduction of the concentration of light impurities such as carbon and oxygen (C, O ≤ 0.1%). In consequence the radiated power decreased, especially in the divertor [5] (see Fig. 1), and the thermal load on the W-coated divertor tiles reached values of over 10 MW/m^2. At these high heat loads it came at several tiles in the LFS divertor (200 μm coatings) to delamination effects. As a consequence all discharges with high heating power (> 7.5MW) were only conducted with active cooling of the divertor plasma by enhancing the radiation with N\textsubscript{2} seeding. The amount of puffed N\textsubscript{2} is feedback controlled by requesting a certain divertor temperature. The latter is estimated by measuring thermoelectric currents in the divertor. Fig. 1 shows the increase of divertor radiation with nitrogen seeding, which reaches power densities comparable with carbon dominated machine conditions. The N seeding does obviously not result in enhanced Z_{eff} values.

For improved H-mode discharges the boronisation resulted in a ~10% reduced performance (stored energy, H_{98} and beta), which is partly explained by less peaked density profiles. With boronisation stable discharges without deuterium fuelling (achieving lower density and collisionality) are obtained again, as the tungsten influxes from the outer limiter are (temporarily) suppressed. As expected the confinement increased by 5-10% and the observed reduced confinement level at higher deuterium fuelling rates could be partly compensated. As a positive surprise it turned out that N\textsubscript{2} seeding does not only protect the coatings of divertor tiles, but also significantly improves the performance of discharges. This is demonstrated in Fig. 5 for an improved H-mode discharge (1 MA, 2.5 T) where the energy confinement increased to H_{98}-factor of 1.25 and the β_N reached 2.7. The achieved plasma energy of 1 MJ is shown in Fig. 4a for comparison. In Fig. 4b the H_{98}-factors are displayed against the gas
fuelling rate for the improved H-mode discharges in a full W wall without boronization and under boronized conditions together with nitrogen seeding. The performance improvement with nitrogen seeding holds for all fuelling rates and leads to energy confinements exceeding thereby the best values which have been obtained in a carbon-dominated machine in the years 2002 – 2006. In line with results from previous campaigns [24], a higher confinement is achieved even with high densities and unboronized conditions at higher triangular plasma shapes. Combining both positive ingredients, high triangularity (up to 0.45 available with restored generator capabilities) and N seeding in the boronised W wall, in the next operation campaign offers new performance perspectives.

5. ITER breakdown and ramp-up scenario studies

The ASDEX Upgrade programme in 2008 also addressed two immediate ITER research needs in view of its operation phase. The LH transition in helium plasmas was studied and a H plasma study will follow in autumn. The results show that the H-mode threshold in helium is hardly different from the one in deuterium both for ECRH and H beam heating and possesses a pronounced minimum at line-averaged densities of $4.5 \cdot 10^{19} \text{ m}^{-3}$. This is of great importance for the first operating phase of ITER without radioactive activation to exploit the possibility for an early H-mode operation phase [26].

A detailed study of an ITER like plasma breakdown and subsequent current ramp-up to $q_{95}=3$ was performed [27], 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 $E\sim 0.25 \text{ V/m}$. ITER plans to use 0.32 V/m. ECRH was used for pre-ionisation at the 2$^{nd}$ 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 ($\leq 1 \text{ MW}$) 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 radiation. 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 ($<3\times10^{19}$ m$^{-3}$) with good plasma performance even with $E<0.3$ V/m (Fig. 6 left). 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 0.71 to 0.97 at fixed plasma current rise rate $dI_p/dt=0.66$MA/s (Fig. 6 right). This is within the foreseen operational boundaries ($l_i=0.7-1.0$) for 15MA operation in ITER given mainly from vertical position control and the OH flux limits. A prerequisite for this low $l_i$ ramp-up is the early transition to a full bore plasma shape, where the breakdown occurs close to the HFS limiter, followed by an LFS limiter ramp-up and an early X-point formation at about 0.5 s. An even earlier divertor transition will be possible with our full generator power in the next campaign. The ramp-down is done in an inverse sequence and by avoiding both an early H→L back transition using additional heating and an inboard limiter contact.

6. 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, but 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 [28]) 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 ($\rho_{tor}=0.2$) inside of the plasma is important to enhance the outward impurity transport and to control the accumulation of tungsten. The ECRH system at ASDEX Upgrade provides this flexibility, but was limited in

![Figure 6: ECRH start and ramp-up assist at AUG.](image-url)

Left: The minimum required electrical field on axis is reduced to ~0.2V/m using 0.3-0.9 MW ECRH (X2).
Right: Internal inductance during current ramp-up for different heating powers and confinement modes, resp.
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, βN~1.6-2) have been demonstrated, keeping W concentrations below 3.10^{-5}.

Improved H-modes in the unboronized W device with H_{98}~1.1 and βN>2.5 have been achieved. After boronization as a positive surprise it turned out that actively controlled N2 seeding does not only replace the missing divertor radiation from light impurities and thus protect the coatings of divertor tiles, but also considerably improves the performance of discharges. The energy confinement increased clearly to H-factors of 1.25 and exceeded thereby the best values, which have been obtained in a carbon-dominated machine in the years 2002 - 2006. Preliminary investigations show that this improvement is due to higher temperatures rather than to peaking of the electron density profile. The change to a fishbone dominated core MHD behaviour might also contribute. These AUG results give added confidence that a tungsten wall is compatible with baseline and advanced operation in future fusion reactors.

Finally, a detailed study of an ITER like plasma breakdown and current ramp-up with low loop voltage using ECH assist for breakdown was conducted with a full tungsten wall. The latter showed reliable ECH breakdown in the ITER scenario without a need for dedicated gyrotrons (170 GHz in O1 mode / 5.2 T or X2 mode/ 2.6 T) and ramp-up at low electric field of 0.25 V/m (ITER design value: 0.33 V/m). The experiments on AUG were confirmed by similar ones at JET and DIII-D and together they form a reliable basis for the plasma ramp-up scenarios planned at ITER.

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