# Operation of ASDEX Upgrade with Tungsten coated Walls

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**Abstract:** Since 1998 a step by step approach to investigate high-Z material at the main chamber walls of the ASDEX Upgrade divertor tokamak was followed, resulting in 7.1 $m^2$ of tungsten coated tiles at the central column during the 2001-02 campaign. Despite this large area, plasma operation was not hampered in any way by tungsten radiation. Results obtained from a variety of confinement regimes indicate that the core tungsten concentration depends mostly on core transport rather than on the tungsten erosion source. For medium density H-mode discharges tungsten concentrations of $\approx 1 \cdot 10^{-6}$ are found. Higher concentrations are observed only under discharge conditions where neoclassical accumulation becomes dominant as in case of strong background plasma peaking. On the other hand, core accumulation can be effectively controlled without noticeable confinement degradation by applying central heating. Unexpected high average tungsten erosion due to ions could be attributed to transient limiter phases, especially during plasma ramp-down.

## Introduction

During the last years it became evident, that the wall problem of a future fusion reactor is still unsolved. The actual design of ITER [1] uses a mixture of beryllium in the main chamber, tungsten at the divertor and carbon at the strike points. Beryllium and carbon suffer from high erosion rate, which will restrict the lifetime of the wall dramatically. Furthermore, the use of carbon has to be restricted due to tritium co-deposition [2]. Alternatives for low-Z materials are high-Z materials like tungsten. Tungsten offers low erosion yields, excellent thermo-mechanical properties, and small tritium retention, but the maximal tolerable concentration in the plasma core is strongly restricted.

Today the experience with high-Z materials in fusion devices is still poor. Experiments are mostly done in test devices and small tokamaks. From the major divertor experiments only ALCATOR C-mod uses molybdenum [3], and tungsten had been applied in the divertor I of ASDEX Upgrade without major problems [4,5,6]. But while high-Z materials may be unproblematic in a cold divertor plasma environment some distance away from the hot plasma column, the same materials used in the main chamber are supposed to be more critical. In addition, if different wall materials are used, material-mixing effects will occur, which can only be tested in a real tokamak. Therefore, based on the experience and diagnostics developed during the tungsten divertor campaign, a step-by-step approach to insert tungsten at the central column of ASDEX Upgrade was started in 1998 [7].

## Strategy to install tungsten at the main chamber

The lack of knowledge of high-Z materials at divertor tokamaks and the disastrous behaviour in limiter tokamaks required acting with caution. As first step an in-vessel coating with a mid-Z material of only a few nm thickness was chosen. For this purpose a siliconisation instead of a boronisation was applied [8]. The plasma performance was not effected by the silicon radiation, and, indeed, low core silicon concentrations of $10^{-3}$ were found in line with previous estimations. This experiment gave confidence to use high-Z material at the main chamber. The heat shield, which covers the central column of ASDEX Upgrade, was chosen for these experiments, as it presents the largest closed surface in the main chamber. As for the divertor experiment, tungsten was selected as high-Z material. In this second step the two lowest tile rows of the central column (Fig. 1) have been coated. This region is predicted

to suffer the highest erosion by CX neutrals [9]. Although 1.2 $m^2$ of W surface was inserted, it had been mostly not possible to detect radiation caused by tungsten [10]. Again no negative influence on the plasma performance was found. From post-mortem analysis the erosion of the tiles was measured about ten times larger than expected from the CX neutral flux. The tile erosion pattern observed indicated ion bombardment as main erosion process [11], [12]. The third step had been a complete coverage of the heat shield, except regions, which are used as limiter during plasma start-up and other limiter scenarios and the areas hit by NBI shine trough. During this phase 5.5 $m^2$ of tungsten tiles had been exposed.

The normal conditioning procedure after a major vessel vent includes wall coating by either boronisation or siliconisation. To investigate the effect of pure, uncoated tungsten surfaces a modified start-up sequence was followed at the beginning of phase three. A silicon layer on unchanged wall parts applied just before venting, was re-activated by He glow discharge. In combination with the cryo-pump plasma restart was possible without deposits on the newly installed tungsten surfaces. Due to the lack of an active getter at the wall the oxygen content of these discharges was significantly higher than for a well-conditioned apparatus [13]. Nevertheless, operation of all relevant scenarios could be investigated. The measured core tungsten con-
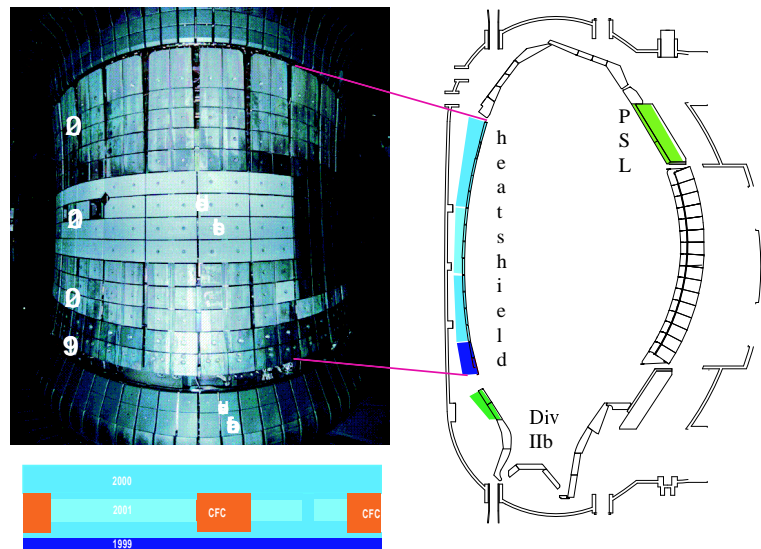


Figure 1: *Overview of the installed tungsten tiles during the different phases. The left picture shows a view on the central column with the different types of tiles. The right figure indicates the positions of W tiles in a poloidal cross section of ASDEX Upgrade. Additionally, the tiles to be installed for the next campaign are indicated. The insert shows the unrolled central column for the 3 steps.*

centrations of a few times $10^{-6}$ had no bad influence on the plasma. Post-mortem analysis of the tiles indicates that a significant part at the heat shield is erosion dominated, i.e. shows pure tungsten surfaces even after a complete campaign with three wall coatings applied. Due to this observation the normal wall conditioning was applied for later investigations. Based on the experience of these phases the shape of the tiles was changed and during the fourth step the limiter region was covered by tungsten tiles. During this 2001-02 campaign 87 % or 7.1 $m^2$ of tungsten had been used at the central column.

**The Tungsten coating**

From test tiles erosion at the central column of ASDEX Upgrade of some 100 nm each experimental campaign is expected [11]. This low erosion rate enables the use of coatings on graphite instead of pure tungsten to test the influence on the plasma [7]. The thickness of the coating is determined by surface roughness of the used graphite. The coating must be thick enough to cover the whole graphite, but thin enough to reduce mechanical stress within the coating. From test tiles and laboratory measurements it was found that a thickness of $\approx$ 1 $\mu m$ would fulfil our requirements best. Before coating the graphite tiles were machined to reduce the surface rough-

ness. Whereas the tiles of the second step had been produced in-house, a commercial manufacturer must produce the large mount of tiles to cover the whole central column. Several coatings of two companies had been produced and tested [14]. From these tests a 1 $\mu m$ thick PVD coating by Plansee was selected. They withstand up to 40 $MW/m^2$ for 300 ms without damage at an ion beam facility [15]. This fits even for the NBI dump at ASDEX Upgrade. Before installation the tiles were characterised using SEM and ion beam techniques. A SEM picture of an erosion-dominated part of a tile after two experimental campaigns is shown in Fig. 2. The surface consists almost of pure tungsten indicating that the coating is still in good order. Note the fine structure of the erosion in the preferential ion sputter direction as indicated by arrow.

The mounting of the tiles at the heat shield was not designed to take the heat during extended limiter phases. In the latter case, thermal loads could cause a slightly movement of the tiles, resulting in plasma exposure and erosion of leading edges. To avoid this effect, a trapezoidal shape of the tiles was chosen, which hides the edges, but also reduces the effective area. So the maximal heat load for limiter operation was still limited. To reduce the gaps in between tiles and minimize thermal movement, a double tile with a parabolic shape was tested. After operation during the 2001-02 campaign no edge erosion was observed on these tiles, indicating that the coating and the modified tile shape are suitable even for some limiter operation. Consequently this new design will be used to cover the whole heat shield as the upcoming 5th step.
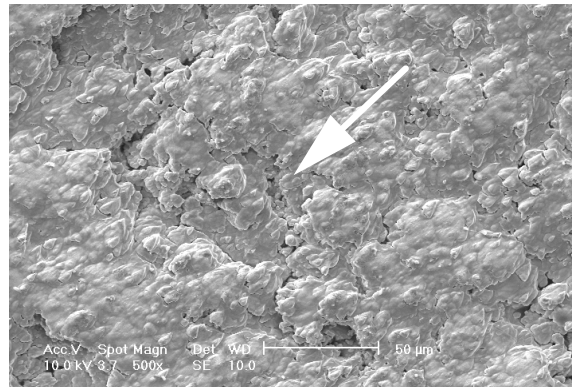


Figure 2: *SEM picture showing erosion dominated region of a tungsten tile at the end of the campaign. The surface almost consists of pure tungsten. The coating covers the whole surface and no deposition is observed.*

**Plasma discharge behaviour**

The operation of ASDEX Upgrade was not hampered in any way by W radiation during all phases. The last two campaigns lasted for 6900 seconds of to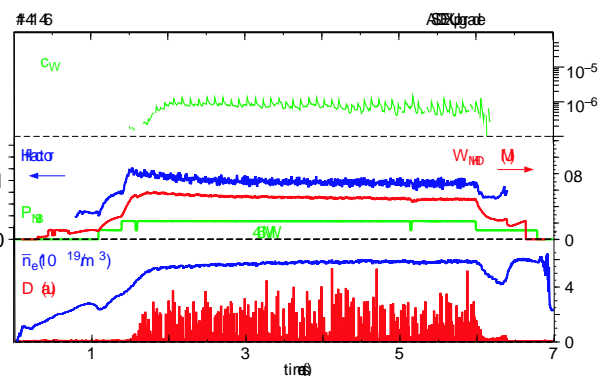tal plasma operation without any damage of the W coating. All types of plasma discharges covering most actual fields of fusion research were carried out [16]. Low triangular ($\delta \approx 0.15$) H-mode discharges are the most common operation regime of ASDEX Upgrade. A typical low density H-mode discharge is shown in Fig. 3. The discharge shows good performance ($H_{ITER92P_{ELMy}} \approx 1$) and a very stable behaviour of all discharge parameters. The tungsten concentration $C_W$ extracted from the central line emission of $W^{46+}$ stays constant at $1 \cdot 10^{-6}$. Spikes on the $C_W$ signal are artefacts due to the influence of saw-teeth on the data evaluation. Detailed analysis of different plasma scenarios is reported by Neu



Figure 3: *Typical medium density neutral injection heated H-mode discharge ($I_p = 1MA, B_T = -2.5T$). The discharge shows good confinement and very stable behaviour.*

et al. [17]. The tungsten core concentration depends strongly on the type of discharge. For discharges with high-density ($n_e/n_{gw} \approx 0.75$), low-$\delta$, all heating powers no tungsten within the detection limit is found. Discharges with medium triangularity show higher energy and particle confinement and a tendency to profile peaking resulting in tungsten accumulation up to $C_W \approx 10^{-5}$. Another important field are enhanced confinement scenarios like improved H-mode, electron ITB and high-$\beta$ discharges. These types of discharges show also enhanced tungsten concentration up to $C_W \approx 10^{-5}$. In general higher concentrations correspond to low plasma densities, hot SOL conditions, scenarios with high particle confinement and non-central heating. Limiter discharges were performed to preserve L-mode discharges with NBI heating. Even for 4.9 MW heating tungsten concentration below $1 \cdot 10^{-5}$ were measured. The highest concentration was found for limiter ITB discharges, which show tungsten concentrations up to $3 \cdot 10^{-4}$. Nevertheless, the discharge survived and after the ITB phase the tungsten concentration dropped to low values again, though the plasma edge remained unchanged [18]. Except for limiter discharges the core concentration found is always well below the maximal tolerable concentration for ITER of $2 \cdot 10^{-5}$. These experiments demonstrate the compatibility of fusion plasmas with tungsten plasma facing components under reactor relevant conditions for normal H-mode discharges. The carbon radiation is not strongly influenced by the tungsten coating. Spectroscopic measurements indicate that even on the tungsten heat shield carbon radiation dominates. In contrast to the previous campaign the plasma was started at the tungsten coated heat shield as limiter. During ramp-up the flux consumption and radiation was slightly enhanced, but not as much as for siliconisation. After transition to divertor operation the tungsten content diminishes again to values as found without tungsten coated start limiter [17].

**Tungsten erosion processes**

Direct spectroscopic measurement of the tungsten influx is only possible during special discharges, because the WI brightness is normally too low to detect. To investigate the erosion of tungsten at the central column a complete row of 14 tiles was polished and coated with only 60 nm of tungsten. The tungsten layer thickness was chosen such as to match the expected erosion rate and enable this way an accurate measurement. The layer thickness was determined before installation and after the campaign using ion beam techniques [19]. Although the tungsten concentrations found in the plasma are quite small, the total erosion of the tungsten tiles is much higher than expected from CX erosion calculations. Whereas deposition was found at the trapezoidal shade positions, erosion mostly occurred at the leading edge of the tile [20,21] (Fig. 4). An ion, which has to follow the magnetic field lines, hits the surface of the tile with a small angle resulting in shadowing, whereas CX neutrals should cause a homogeneous erosion pattern. Therefore, since the erosion varies strongly in space, this clearly points to ion sputtering. Especially at the
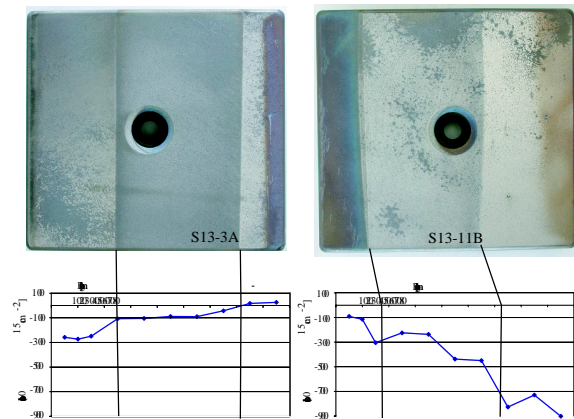


Figure 4: *Tungsten erosion pattern. The erosion is locally variable due to the imperfect mounting of the tiles. At the left side of the column the main erosion occurs above the midplane, whereas below the midplane the right side of the tile is eroded most.*

mounting of the tile a sharp transition from erosion to deposition dominated regions is ob-

served. In Fig. 4 two tungsten tiles from the upper and lower part of heat shield are shown after exposure. Whereas the main erosion at the upper tile is on the left side the, at the lower tile the right side is eroded. This coincides with the ion drift direction of a limiter plasma.

A major part of the eroded tungsten should be promptly deposited within a gryo radius [22]. This effect causes a movement of the eroded tungsten into the deposition dominated areas. Whereas ion beam measurements suffer from the tungsten layer below the deposit, SIMS and AES data indicate that the deposition layer mainly consists of low-Z materials: carbon and boron. Only small amounts of tungsten are found at these layers. The deposition-dominated areas are comparable to the erosion dominated ones. Balancing the eroded tungsten no evidence is found, that a major part of the eroded W is deposited on the heat shield tiles. During divertor operation most of the material in the SOL should be deposited near the strike points. A poloidal tile row of the divertor was removed and analysed using ion beam techniques. For divertor II a high tungsten deposition was found at a horizontal structure just below the heat shield. This structure has been inclined and is about parallel to magnetic flux surfaces in Div IIb; only slight deposition is now found at this location [23]. For both divertors the tungsten deposition near the strike points is very small. This observation is different to other impurities as C and B, which are mostly found near the inner strike point. Summing up all tungsten deposits measured in Div IIb less than $\approx 10\%$ of the eroded material is detected.

To identify the main erosion processes Langmuir probes were installed at the central column in-between the tungsten coated tiles. Using this probes as single swept probes, clear characteristics could be observed. The data were evaluated by the double probe model [24] yielding the plasma parameters close to the tungsten surface. Data of a typical H-mode shot are shown in Fig. 5. Three phases during a discharge can be distinguished: start-up limiter, divertor and ramp-down limiter operation. During the start-up hot plasma is observed. After the transition to the divertor phase low-density plasma with moderate temperatures is found. During the ramp-down of the plasma (7 - 8 sec) rather dense, but moderately hot plasma is observed. To avoid disruptions auxiliary heating is applied also during plasma ramp-down. The erosion of tungsten is known from the former tungsten divertor experiments to be dominated by impurities. In ASDEX Upgrade SOL typically 1 % of carbon is present dominating impurity. Whereas during the start-up phase the impurity concentration may be lower, this value seems reasonable also for the ramp-down plasma. Appling an erosion model [25] using the measured plasma parameters, the erosion during these three phases can be calculated.
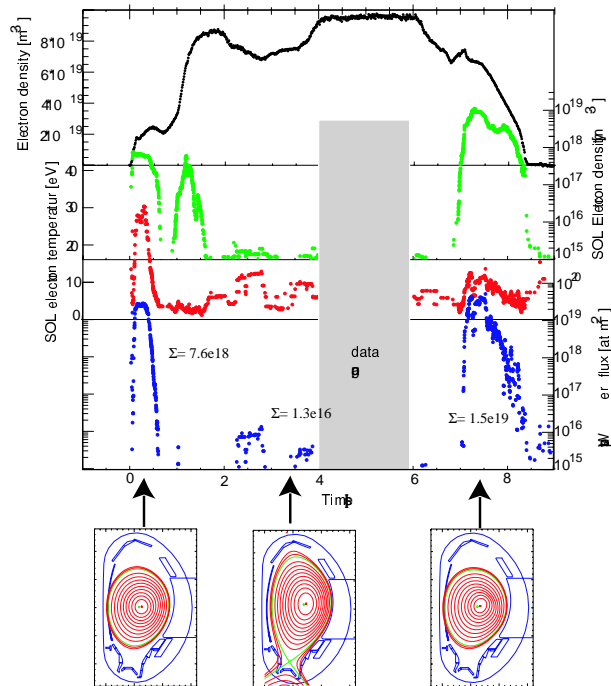


Figure 5: *Plasma properties at the heat shield derived from Langmuir probe data for a typical discharge. Using an erosion model the tungsten erosion can be estimated. The integral values for plasma ramp-up, flattop and ramp-down are also indicated. At the bottom magnetic contours for this three phase are shown.*

These calculations reveal that the tungsten erosion mainly occurs during plasma ramp-down. The enhanced tungsten erosion during ramp-up and down is, however, still to be confirmed by spectroscopic measurements [17].

Taking into account that the mounting of the tiles results in different angles of magnetic field incidence and a strong variation of the erosion rate (and also that the database of the langmuir probe data is still limited), the derived erosion is in agreement with the post-mortem analysis. Due to the shape of the plasma during ramp-down the main direction of erosion as indicated in Fig. 4 can be explained. Since there is no x-point during ramp-down a low deposition in the divertor is expected. However, although the main erosion during plasma ramp-down is consistent with the observed erosion and deposition pattern, the main deposition area of tungsten is still not identified. During flattop the derived erosion rate is more than three orders of magnitude lower. For long-term discharges only the behaviour during flattop is relevant. DIVIMP code simulations have been started to analyse the transport during the flattop phase [26]. In preparation of the use of tungsten at other regions of ASDEX Upgrade, test tiles were installed at the low field side and in the divertor region. These indicate that erosion by arcs might play a role at the transition region from the divertor to the inner heat shield.

**Tungsten transport into the core plasma**

To extrapolate the observed tungsten concentrations to future fusion devices one has to understand the mechanism, which determines the core tungsten concentration. The presence of large tungsten areas at the main chamber enables realistic measurements of tungsten transport in a fusion device. At ASDEX Upgrade the core tungsten concentration varies strongly for discharges with same plasma edge conditions, a fact which indicates that the transport of tungsten in the main plasma plays an important role. For scenarios, which use NBI injection with reduced beam voltage, i.e. non central heating, even accumulation can be observed. This effect was studied in detail by a variation of the heating profile, without modification of the scrape-off layer plasma. It was found that central heating by ECRH or ICRH could reduce the tungsten core concentration significantly. Obviously the core tungsten concentration depends (not only on the tungsten source but also) mainly on the trans-



Figure 6: *Standard H-mode discharge, which is performed each day, showing all the details of the tungsten transport in ASDEX Upgrade.*

port of the material from the scrape-off layer into the core plasma. The wave heating experiments indicate that external knobs are available, which allow controlling the core concentration of tungsten. Detailed studies on tungsten transport at ASDEX Upgrade were presented by Neu [17] and Dux [27]. Fig. 6 summarizes the behaviour of the tungsten core concentration during the H-mode standard discharge, which shows most of the details of the tungsten transport. The X-ray data reflect the core tungsten, whereas the pedestal region dominates the VUV data for flattop plasma parameters. During start-up enhanced core tungsten is found as expected for limiter operation. The tungsten concentration decreases by a factor of ten after the transition to the divertor regime, reflecting the change of the W
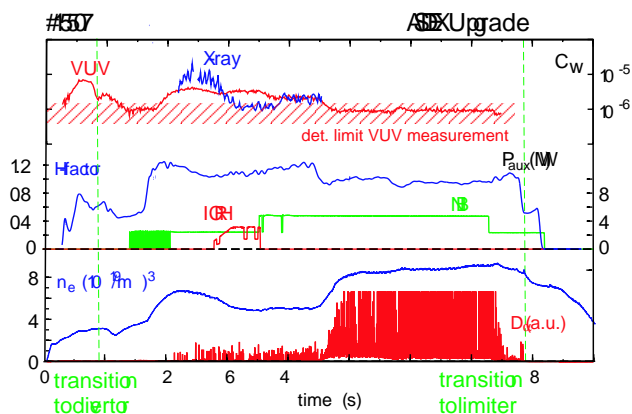
source. Transition to H-mode at low heating power leads to good particle confinement and peaked tungsten profiles, which is inverted by central ICRH heating. Enhancement of the plasma density reduces the tungsten concentration below the detection limit. Unfortunally the ramp-down phase was not recorded for this discharge to due restriction of data acquisition. Only in a minor part of all discharges accumulation was observed, but always for all impurities, not only for tungsten. These discharges are characterised by moderate heating were the anomalous diffusion is low [28]. Only the most central channels of the soft X-ray cameras observe enhanced radiation during accumulation indicating that only a small volume is affected. Due to this fact usually the total radiation is not changed and no degradation of the total confinement is observed. This behaviour can be understood in terms of neo-classical transport [27,28]. Although transport calculations require accurate measurement of the hydrogen and impurity density as well as temperature profiles, which are not routinely available.

Central heating provides an effective tool to control the tungsten concentration, but may influence the confinement. Detailed investigations were done using core ECRH in improved H-mode discharges (Fig. 7). These medium triangularity discharges show very good energy confinement ($H_{ITER-H98y} \approx 1.3$), but are affected by continuously rising W-concentration. Appling central 1.2 MW ECR heating can reduce the W concentration by a factor of 30, but the confinement is also degraded. ECRH with only 0.4 MW did not affect the W-concentrations. The best behaviour is observed by a short



Figure 7: *Tungsten concentration $C_W$ for similar discharges with different central ECRH heating. The W concentration can be controlled without degradation of the discharge (15524).*

1.2 MW ECR preheat and a reduction to 0.8 MW thereafter. The W-concentration was reduced as long as the central heating remained, and no degeneration of the confinement was observed.

**Summary and outlook**

Since 1998 a step-by-step approach to increase the tungsten area at the main chamber walls of ASDEX Upgrade was pursued. During the last campaign 85 % of the central column or 7.1 $m^2$ of W coated tiles were used. Graphite tiles with 1 $\mu m$ of tungsten coating are sufficient for present experiments.

The erosion rate and pattern on the tiles hint to sputtering by ions as main erosion mechanism. The ion fluxes are confirmed by Langmuir probe measurements at the heat shield showing that the main erosion happens during ramp up and ramp down. Only weak deposition is found at the divertor tiles, reflecting much lower erosion during flattop. The plasma operation of ASDEX Upgrade was not negatively influenced by the tungsten radiation. Central tungsten concentration are normally in the range $\approx 10^{-6}$. Higher concentrations up to $10^{-4}$ are observed only under discharge conditions where neoclassical accumulation becomes dominant as in case of strong background plasma peaking.

The central tungsten concentration is mostly connected to the transport into the core plasma and less strongly to the tungsten source at the scrape off layer. After start-up at a tungsten limiter, the
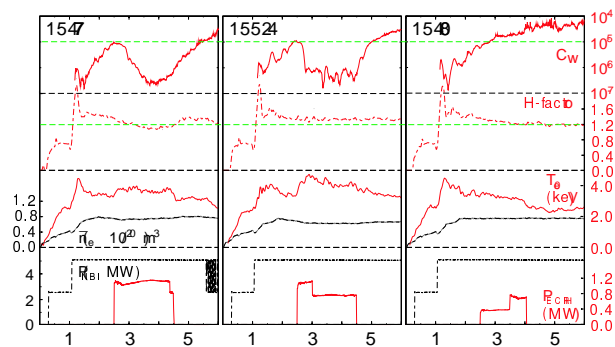
concentration decreases rapidly after x-point formation. The peaking of the W density occurs very localized for $\rho_{pol} < 0.2$. Central heating provides an effective tool to reduce the tungsten core concentration, without strongly affecting the global energy confinement. Quantitatively this can be understood in terms of neoclassical impurity transport. These ASDEX Upgrade results are very encouraging. Especially the required heating profile to prevent accumulation seems to be consistent with the ITER expectations.

To continue the tungsten experiments, new tiles with the optimised shape will be inserted at the heat shield during the 2002 opening. Since protection tiles at the outer side have to be replaced due to mechanical reasons, the new tiles will also be covered with tungsten. Additionally the inner retention module of the divertor, which represents the ITER baffle position, will be covered by tungsten. In total 15 $m^2$ or 40 % of the total SOL surface will be covered by tungsten for the next campaign. Only the lower and upper divertor and the protection limiters at the low field side will remain as carbon. This offers the investigation of fresh surfaces and may lead to a substantial suppression of the C content. The reduction of low-Z material is necessary to investigate SOL without low-Z radiation, which up to now dominates the divertor power balance.

To study the tungsten erosion a set of 5 Langmuir probes will be operated at the heat shield. Additional test samples for the application of tungsten at the low field side protection limiters will be tested with the midplane manipulator.

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