

Ten years of W programme in ASDEX Upgrade - Challenges and Conclusions

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Abstract. Since 1999 ASDEX Upgrade increased its tungsten plasma facing components and finally reached a full W coverage in 2007. Most of the initial goals of the investigations were successfully achieved. A highlight of the investigations were multiple start-ups and operation without any boronisation demonstrating that performance and confinement similar to boronised operation with carbon PFCs can be reached in high power, high density discharges. This also allowed the investigation of the hydrogen retention without disturbing effects from the low-Z coating. A strong reduction of hydrogen retention was found in gas balance measurements as well as in post mortem analyses. On the other hand, an almost complete suppression of low-Z divertor radiation was achieved after boronisation, providing valuable information on the control requirements of radiative cooling by artificially introduced impurities. Among the challenges remains the strong increase of the W source and W concentration resulting from ICRH. At the same time it helped to identify the underlying physics and may lead to solutions superior to the presently used ones.

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1. Introduction

Tungsten is the top candidate for the plasma facing (armour) material in future fusion reactors, due to its capability to survive in high temperature, high neutron irradiation environment. Nevertheless the fusion community only reluctantly uses it at plasma facing components (PFCs). The reason can be found in its strong ability to hamper plasma operation as found in early W limiter tokamak experiments (see for example [1]) and - on the other side - the very beneficial behaviour of carbon based PFCs. Also ASDEX Upgrade applied W as plasma facing material (PFM) very cautiously. Ten years ago the first W coated graphite tiles were installed at the central column. This part was identified to allow first investigations with W PFCs in the main chamber, but was also considered not to hamper the usual operation [2]. Previously, in 1996 the strike point area was equipped with W coated tiles, demonstrating that the use of tungsten is feasible in a divertor tokamak. At the same time strong carbon deposition at the inner divertor was observed making evident that the main chamber carbon PFCs are a significant source of impurities [3,4]. From 1999 on, the area of W components was continuously increased until finally in 2007 100% W coverage was reached [5, 6], representing the only full tungsten fusion device. This step by step approach not only allowed to identify the role of different local W sources for the W density in the plasma and to investigate the effect of mixed materials, but also to adjust the operational procedures to a narrowing operational space. It should be pointed out, that the majority of the recent results were achieved under unboronised conditions, an attempt which was performed with the same stringency only in Alcator C-Mod [7].

This contribution intends to highlight the major achievements of the programme so far and will then concentrate on more recent results concerning details of the operational experiences. In the last part, remaining issues to be solved on the way to ITER and Demo will be uncovered and the attempts undertaken in ASDEX Upgrade will be sketched.

2. Technical realisation of W PFCs in ASDEX Upgrade

Tungsten coatings on fine grain graphite have been used at ASDEX Upgrade in order to provide a solution which complies with the technical boundary conditions when transforming from C to W PFCs. For the W divertor experiment in 1996 'thick' W vacuum plasma spray (VPS) coatings had been chosen [8, 9]. For the coatings in the main chamber physical vapour deposition (PVD) coatings with various thicknesses up to 4 μm were employed because of their high power load capability [10], benign failure mechanism and their lower price compared to VPS coatings. The transformation to a full W device was completed in 2007, when the strike point tiles of the lower divertor were coated. Since the erosion in the outer divertor can exceed 1 μm per campaign [11], it was decided to use again VPS (200 μm W) coatings at this location. In order to avoid cracks, graphite with adjusted thermal expansion was used (SGL R6710, $\alpha^{R6710} \approx 5.8 \cdot 10^{-6} \text{ K}^{-1}$) and comprehensive thermal screening and cyclic loading tests were performed before installation [12]. Nevertheless, the VPS coating of two tiles (out of 128) delaminated during the 2007/2008 campaign after the first application of boronisation (see below) in discharges with heating powers above 14 MW and target power loads in excess of 10 MWm^{-2} . These damages led to large area delamination and ejection of macroscopic particles into the plasma, eventually causing disruptions. The damage was repaired in an unscheduled vent and a post mortem analysis of the tiles revealed that the coating did not fail at the tile edges but at the central part. Additionally, it became evident from loosely bound remains of the coating that the coatings delaminated as a whole obviously due to adhesion problems already existent from the production process. This defect pattern led to the injection of the large 'W flakes' (several mm^2) into the divertor plasma, which was observed during the operation. After the repair of the damaged tiles, operation with similarly high heating power was resumed, which again led to a similar coating failure. However, this time the machine was not vented and the machine was operated with the damaged tile. The initially released large coating fragments were successfully removed from direct plasma contact by a few disruptions and operation could be continued with reduced performance

(about 300 discharges up to 10 MW heating) until the next scheduled vent. After the repair of all damaged tiles, routine radiation cooling in discharges with auxiliary heating power above 7 MW allowed to perform high power discharges without further damage. Details of the recent coating performance are presented in [13]. Despite the success achieved by routinely using radiation cooling, it was decided to exchange all VPS coated tiles to PVD coated ones during the last vent, in order to reduce the risk of hampering the operation by injection of parts of a damaged coating. The new PVD coating consists of 10 μm W on a 3 μm Mo interlayer, similar to the one adopted for the JET ITER-like wall project [14], but using fine grain graphite as substrate instead of CFC. Figure 1 presents a view into the vacuum vessel of ASDEX Upgrade after the first campaign with a complete W coverage of the PFCs. This first campaign was performed without any boronisation and, as can be inferred from the shiny surfaces, there are almost no thick deposits found except for the inner divertor [15]. The tiles of one sector at the inner divertor were exchanged after the campaign for surface analysis (central part of the picture).

3. Recent achievements of the W programme in ASDEX Upgrade

The use of W as a PFM is motivated by the need of low tritium retention, low erosion and stability against neutron irradiation. On the other hand, the following questions arise: Is reliable tokamak operation possible with high-Z PFCs, are they compatible with standard and advanced H-mode scenarios and with the available heating methods? In order to tackle these questions in the full W ASDEX Upgrade the major part of the last two experimental campaigns was operated with new/cleaned W surfaces and without boronisation.

A major challenge of operating without boronisation is the start-up of the device after a vent, since the available conditioning methods at ASDEX Upgrade (baking only up to 150° C and operation with PFCs at room temperature) do not allow to get efficiently rid of adsorbed water and oxygen (see for example [16]). Besides oxygen, carbon still was observed in the plasma with concentrations in the range of 1 %. The carbon source is not conclusively identified, but chemical erosion by D and O from old co-deposits on the stainless steel vacuum

vessel wall is seen as a strong candidate [17].

In Fig. 2 the startup sequence during restart in September 2008 is shown. Before, three restarts without any boronisation were performed and also this restart was done without prior boronisation (in the campaign before three boronisation were performed, but the layers were wiped wet during the vent). As can be judged from the figure, a very quick recovery of the full W device could be achieved using an optimised start-up sequence. ECRH is added quite early in the discharge to increase the electron temperature and thereby the conductivity, without increasing the density to stay well below the Greenwald density limit. NBI is added from 0.3 s on to increase further the available heating power. The discharge is ramped up at the low field side limiters which turned out to be conditioned faster than the central column and the transition to the divertor is performed as early as possible ($t = 0.6$ s), which is governed by the voltage available for the poloidal field coils. With this recipe it took only 5 discharges to reach the preprogrammed current flat-top and only four more to achieve the first H-Mode transition.

The deuterium retention was investigated by post-mortem surface analysis [15] as well as in gas balance measurements [18]. Both analysis methods result in a strong reduction (\approx factor 10) compared to the results with C PFCs, in line with expectations from laboratory measurements on the deuterium retention in W. Detailed TDS, NRA and SIMS investigation reveal however, that the diffusion in bulk W is deeper than observed in similar samples from D irradiation in the laboratory [19].

In order to deduce the specific behaviour of W sources and its transport a comprehensive diagnostic of the W influx and the radial W profile is necessary. As an example, figure 3 shows the temporal evolution of some plasma parameters for an ASDEX Upgrade discharge (#23476) with different heating methods. The W-influx is deduced from the W I line at 400.9 nm as described in [20]. The W concentration is deduced from the quasicontinuum emission at 5 nm and the spectral line at 0.794 nm emitted from Ni-like W^{46+} [21]. The first gives the W concentration close to the plasma edge (around $T_e \approx 1$ keV), whereas the latter represents the central concentration (≈ 3 keV) in typical ASDEX Upgrade discharges. Already from

these few time traces a lot of information of the qualitative behaviour of W for different heating methods can be derived. During the first phase (until $t = 2.5$ s) the plasma is heated by neutral beam injection (NBI) only. During this phase the W source at the limiter in the main chamber is more than a factor of ten smaller than the divertor W source. At the same time the W concentration is strongly peaked, as can be seen from the ratio of the central to edge W concentration, which is about 20. This strong peaking - also denoted as accumulation - was already observed in early experiments using W injection in ASDEX Upgrade [22] and test-limiter experiments in TEXTOR [23]. It is usually explained by neoclassical inward drifts, which can be dominant in the absence of large turbulent transport or macroscopic instabilities as sawteeth [24]. However, central (electron) heating strongly increases the anomalous impurity transport [24] and suppresses the accumulation as already shown in [25] and confirmed at other devices [26–28]. At $t = 2.5$ s the NBI is switched off and at the same time the ion cyclotron resonance heating (ICRH) is switched on. Immediately, the limiter source increases by a factor of 10. On a longer timescale - reflecting the 'slow' particle transport within the plasma - the edge W concentration rises (by about a factor of 4), but at the same time the central concentration decreases to a value close to that at the edge. This phase is characterised by dominant anomalous transport, which tends to reduce W density gradients (see above). At the same time it can be concluded that the main chamber (limiter) source has a much larger impact on the edge concentration than the divertor W source, since the sum of both barely changes during this phase. This fact is further illustrated in the third phase of the discharge from $t = 3.5$ s on. There, a part of the NBI is switched on again, resulting in an increased W source in the divertor, which is not reflected in the main chamber concentrations at all. More quantitatively, such experiments yields divertor retention of about 80% to above 95%, depending on the specific discharge conditions.

Similar conclusions are drawn for the behaviour of Mo sources in Alcator C-Mod, but under unboronised conditions there even the confinement is degraded by the strong Mo-influx caused by ICRH [7]. This reduced performance is probably caused by the fact that ICRH is the only major auxiliary heating method at Alcator C-Mod and increasing the total heating

power always leads to a strong increase of the Mo influx.

Concerning the plasma performance in ASDEX Upgrade, it could be shown [29, 30] that similar confinement could be reached with unboronised W PFCs compared to operation with boronised graphite PFCs, although not in the complete operational space. The main ingredients for successful W operation are supplying enough central heating and using a gas puff in the range of several 10^{21} s^{-1} . More quantitative details on the operational space can be found in [17]. To widen the operational window for investigation with collisionalities in the centre closer to those of ITER, boronisation has been performed in the 2008 campaign. This allows to operate at lower edge densities and higher edge temperatures. After boronisation, the C and O concentrations were reduced to $\approx 0.1 \%$ and below, respectively, consequently leading to very low divertor radiation [17]. In the course of these investigations the maximum available power was used, which in turn led to power loads above 10 MWm^{-2} and to the damage of the divertor coatings (see Sec. 2 and [13]). As a consequence all discharges with additional heating above 7 MW were radiatively cooled by nitrogen seeding. The cooling is controlled by the thermo-electric current due to the different plasma temperature at the inner and the outer target, which turned out to be a very robust scheme [31]. Similarly to earlier experiments with Ne seeding [32], a reduction or increase of the W sputtering yield compared to the unseeded phases was observed. In discharges with very high density, divertor plasma temperatures below 5 eV could be achieved leading to a complete suppression of the W sputtering [33]. The surprising side effect of this procedure was that an increased confinement was observed after the injection of nitrogen combined with smaller ELMs (at higher frequency) in almost all discharges. The physics behind this improvement is not yet understood, but detailed profile measurements point to a stabilizing effect in the pedestal, which increases the ion temperature throughout the whole plasma radius [30].

4. Remaining challenges to be investigated in ASDEX Upgrade

In the following a few issues will be presented which will stay in the focus of the W investigations. They are either the response to specific questions arising from ITER or they

are of a more general nature originating from the use of high-Z PFCs.

During current ramp up, ITER will have to operate at densities in the range of $1 \cdot 10^{19} \text{ m}^{-3}$ and using additional heating power of several MWs [34]. Although there is no fully self consistent modelling available yet, very high temperatures at the separatrix are expected. The values adopted in [34] are 25 eV for the ohmic phase, 100 eV for L-mode and 190 eV for H-Mode. These temperatures are close to (L-Mode) and above (H-Mode) the threshold where the self-sputtering yield of W can get above 1 and thereby leading to a catastrophic increase of the W influx. Even though such a behaviour seems to be very implausible, because the strong W influx would cool the plasma and a stable equilibrium eventually could be reached, it might not allow a successful ramp-up because of flux-swing limitations. In order to provide information on this topic, divertor discharges were performed with as low as possible densities and ECRH at the maximum available power of about 2 MW. Until now, line averaged values in the range $1 - 1.5 \cdot 10^{19} \text{ m}^{-3}$ were achieved leading to divertor temperatures of about 50 eV and thereby to sputtering yields of a few 10^{-2} . The central W concentration reached $5 \cdot 10^{-4}$ not posing any problem to the discharge as could be inferred from the stable high central electron temperature ($T_{e0} > 6 \text{ keV}$).

The almost complete suppression of low-Z divertor radiation, which was achieved after boronisation, provided valuable information on the requirements of radiative cooling by artificially introduced impurities [17]. ITER will be in a similar situation after having replaced the CFC at the strike point position by tungsten PFCs: Be is an inefficient radiator and W, although having a high radiation loss parameter, will not provide enough edge and divertor radiation, because of its low concentration allowed in ITER. Since ITER relies on semi-detached operation and radiation cooling in order to keep the power load at or below 10 MW m^{-2} , safe control algorithms have to be developed and the species and/or mixture of the seeding gas has to be optimised for maximum edge radiation with minimised core losses. Research on this issue will be one of the major topics in the ongoing campaign since the reduction of power loads in the divertor will also be crucial.

During the last 2 years operation in ASDEX Upgrade was hampered by the defect of one

out of three flywheel generators leading to limitations in the available power and energy for the power supplies of the coils and the auxiliary heating. From autumn 2009 on the generator will be back in operation and the full range of plasma shapes ($\delta \leq 0.45$), plasma currents ($I_p \leq 1.4$ MA) and heating power ($P_{aux} \leq 28$ MW) will be accessible again with pulse lengths of up to 10 s. An immediate application of the capability for longer pulses will be gas balance investigations, again under unboronised conditions. The experiments done so far show a saturation of the wall after puffing about $5 \cdot 10^{22}$ D atoms, which is typically reached after 3 s. After this initial loading of the wall, the amount of gas retained is zero within the error bars [18]. The extension of the flattop from about 4 s to about 8 s will allow to investigate this observation in more detail.

ECRH is used as a successful standard tool for central heating, because the W influx from the ICRH limiters can lead to too high W influxes during operation of ICRH. However this limits operation close to toroidal fields of $B_t = 2.5$ T and to densities below about $1.2 \cdot 10^{20}$ m⁻³, because of the cut-off in X-Mode operation. To ameliorate these restrictions heating in O2- and X3-Mode have successfully been tested and will be applied more frequently in the future. With the commissioning of two new 1 MW gyrotrons during 2009, there will be also the possibility to combine the different heating schemes.

However, since ICRH would be more flexible, large efforts are undertaken to find ways to reduce the W influx during its operation. This is not only important for ASDEX Upgrade, but also for ITER, where a significant amount of ICRH is foreseen and increased impurity influxes either from the W baffles or from the Be first wall could negatively influence performance and life time. There are approaches to lower the W sputtering operationally as for example by increasing the clearance between plasma and the antennae or by increasing the density, but it seems that a noticeable reduction can only be achieved by a new antenna design, which tries to reduce currents in the antenna box [35]. The final solution will take several years and most probably will be implemented only together with the conducting wall in ASDEX Upgrade, but its concept can be already tested in the 2010 campaign.

As stated in the previous section and described in more detail in [17], ELMs play a crucial

role in suppressing the unduly high W influx through the edge transport barrier. It has been shown earlier [31] that the beneficial influence is also provided by ELMs provoked through pellets which would be in line with the idea of pellet ELM pace making in ITER. These route will be addressed again after the re-commissioning of the fast ELM injection system which is currently under way. Another way to ameliorate the problem of too energetic ELMs in large devices is their complete suppression by resonant magnetic perturbations as pioneered by DIII-D [36, 37] and JET [38, 39]. However, common to these experiments is the tendency that the separatrix temperature increases during the ELM suppression phases which would be counter productive in the case of high-Z PFCs. ASDEX Upgrade will install internal coils during the next year to investigate the compatibility of this promising method with high-Z plasma facing components.

5. Summary and Conclusions

Most of the initial goals of the W programme could be addressed successfully throughout the last ten years. Amongst the highlights of the investigations were the first experimental evidence for the suppression of central W accumulation by central heating [25] and the start-up and operation without any boronisation demonstrating that performance and confinement similar to boronised operation with carbon PFCs can be reached in high power, high density discharges [16, 29]. Among the challenges remain the strong increase of the W source and W concentration resulting from ICRH and the need for rigorous modelling for the extrapolation of the AUG results to ITER. Most of them are currently addressed and results are expected in the upcoming years. Clearly, not all questions posed by ITER can be answered in ASDEX Upgrade, amongst which are the effects of material mixing with Be, the melt behaviour under transients or the change of the hydrogen retention due to damage by high energy neutron irradiation [40]. Some answers may be provided by the ITER-like wall project in JET [41], but others have to be answered by dedicated experiments in other plasma devices or by modelling. However, the results yielded so far do not exclude the use of W in ITER as a standard PFC. Moreover, even the drawbacks revealed so far, as for example the strong W sources induced

by ICRH, helped to identify the underlying physics and may lead to solutions superior to the presently used ones.

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Figure Captions

Fig. 1:

View into the fully W coated ASDEX Upgrade before restart spring 2008 (after one campaign with complete W coverage).

Fig. 2:

Startup sequence during restart in September 2008. The evolution of main plasma parameters and heating trajectories of the first 5 consecutive discharge trials is given. From the top to the bottom the plasma current I_P , the central line averaged density n_e , the W concentration c_W (only for the last discharge of the sequence), the auxiliary heating power from ECRH P_{ECRH} and NBI P_{NBI} are presented. The vertical line marks the transition to the divertor.

Fig. 3:

W influx and W concentration in ASDEX Upgrade during discharge #23476. The two top graphs present the heating power (P_{NBI} , P_{ICRH}) and total radiation (P_{rad}) as well as the line averaged density (n_e) and the stored energy (W_{mhd}) of the plasma. The third graph highlights the W influx (Γ_W) from the limiters and the divertor and the bottom insert shows the deduced W concentration (c_W) at the plasma edge and the centre.

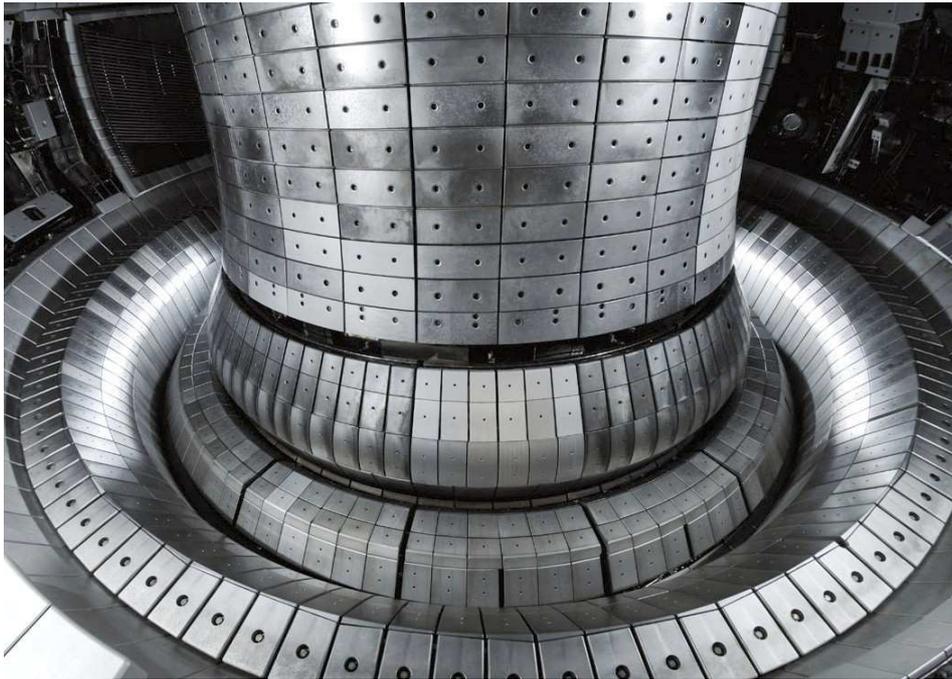


Figure 1.

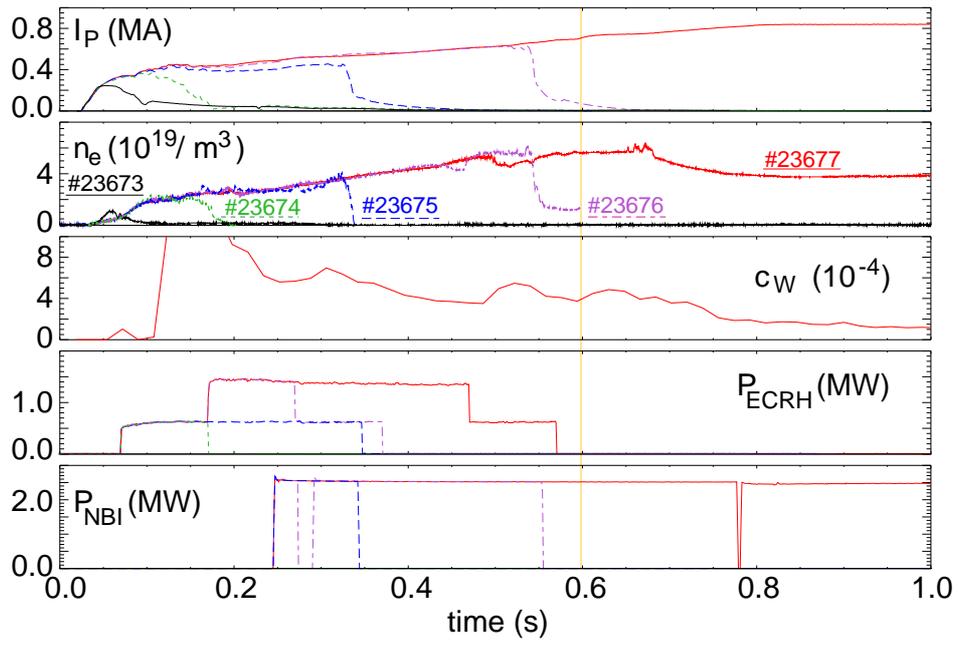


Figure 2.

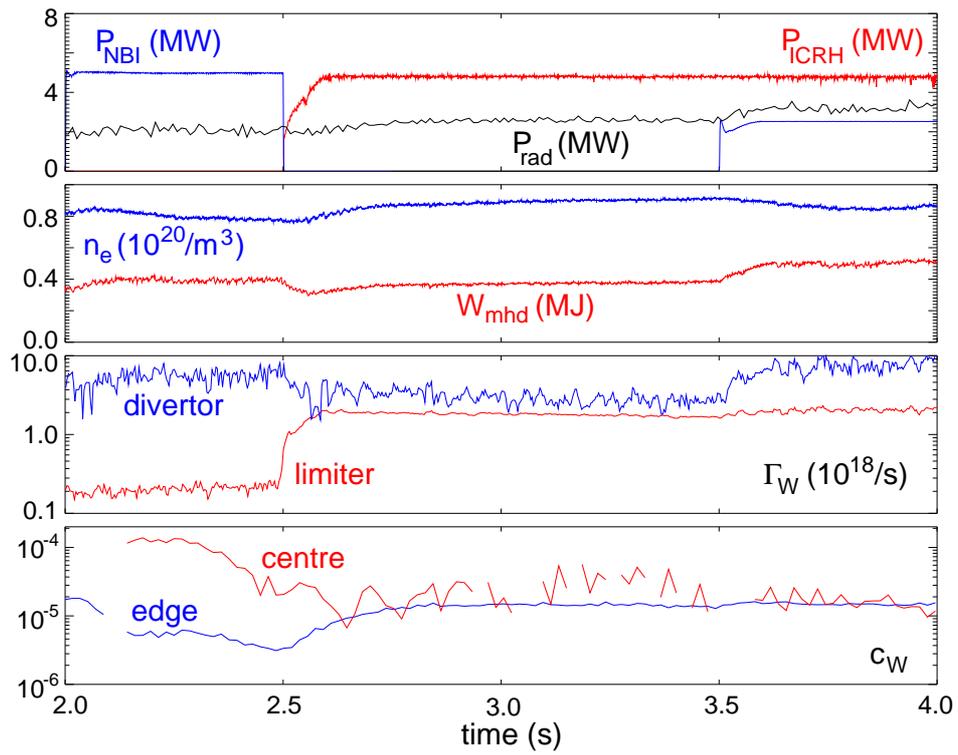


Figure 3.