

On Impurity Handling in High Performance Stellarator/Heliotron Plasmas

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Abstract. The Large Helical Device (LHD) and Wendelstein 7-X (W7-X, under construction) are experiments specially designed to demonstrate long pulse (quasi steady-state) operation, which is an intrinsic property of Stellarators and Heliotrons. Significant progress was made in establishment of high performance plasmas. A crucial point is the increasing impurity confinement towards high density as observed at several machines (TJ-II, W7-AS, LHD) which can lead to impurity accumulation and early pulse termination by radiation collapse at high density. In addition, theoretical predictions for non-axisymmetric configurations prognosticate the absence of impurity screening by ion temperature gradients in standard ion root plasmas. Nevertheless, scenarios were found where impurity accumulation was successfully avoided in LHD and/or W7-AS by the onset of drag forces in the high density and low temperature scrape-off-layer, the generation of magnetic islands at the plasma boundary and to a certain degree also by ELMs, flushing out impurities and reducing the net-impurity influx into the core. Additionally, a reduction of impurity core confinement was observed in the W7-AS High Density H-mode (HDH) regime and by application of sufficient ECRH heating power. The exploration of such purification mechanisms is a demanding task for successful steady-state operation. The impurity transport at the plasma edge/SOL was identified to play a major role for the global impurity behaviour in addition to the core confinement.

1. Introduction

In Stellarators and Heliotrons the plasma confining magnetic fields are exclusively produced by external coil systems without means of net currents induced in the plasma column as being a fundamental part of the tokamak concept. Consequently, these devices do not suffer from current related drawbacks (disruptions, instabilities) and offer the intrinsic potential for continuous plasma operation. The Large Helical Device (LHD) and Wendelstein 7-X (W7-X), the latter presently being under construction, already comply with the technical requirements of long pulse operation (superconducting coils) and focus upon the establishment of high performance long-pulse plasmas. One critical issue is the avoidance of impurity accumulation which – under unfavourable confinement conditions for impurity ions - can lead to early pulse termination by radiation collapse. Theoretical predictions based on neoclassical transport theory for non-axisymmetric configurations underline the importance of this issue: opposite to tokamaks where the "ion temperature screening" of impurity ions is a beneficial mechanism preventing impurity accumulation, for stellarators all driving forces are predicted to principally support accumulation in the standard case with negative radial electric field, the so-called ion root regime [1]. Only in low-density scenarios at W7-AS and LHD positive ion or electron roots can be established as a tool to flush out impurities. On the other hand, even high pressure plasmas with high density and ion temperature simultaneously

achieved within one discharge are interesting candidates for high performance operation but might be afflicted with impurity problems. The relevance of the unfavourable predictions for helical devices finally depends on the predominantly neoclassical or turbulent character of the plasma. However, neoclassical ion transport increases with temperature and might gain more and more of importance in fusion relevant plasmas at high temperatures. For this reason, exploration of favourable scenarios capable of preventing impurity accumulation becomes important. The establishment of proper ambipolar or driven radial electric fields and simultaneously the reduction of impurity influx at the plasma edge is required for long-pulse impurity control. The same issue is of course of relevance for helium exhaust in future devices. In this work an inter-machine comparison of selected purification processes is presented in the frame of the International Stellarator/Heliotron Profile Database (ISHPDB) activities in order to prepare the physics basis for long pulse operation in Stellarators/Heliotrons.

2. Observed Trends for Impurity Confinement

For low density plasmas, impurity transport studies analyzing the temporal and spatial evolution of impurity radiation in response of transient impurity sources (laser blow-off, impurity doped pellet injection etc.) reveal poor impurity confinement in TJ-II and LHD ($n_e < 10^{19} \text{m}^{-3}$) as well as in W7-AS ($n_e < 5 \times 10^{19} \text{m}^{-3}$) [2,3,4]. This was attributed to either strong turbulent transport (W7-AS, LHD [5,7]) or electron root features at low collisionality (LHD [6]). Consistently, the plasma discharges could be maintained without any accumulation of intrinsic impurities.

Additionally, a clear density dependence of the impurity confinement emerges during dedicated density scans in TJ-II, W7-AS and LHD [2,3,4] due to an increasing ratio of inwards convection v to diffusion coefficient D in the bulk plasma and thus longer confinement times at higher density as shown in FIG.1. This was partly attributed to a density dependence of D in W7-AS consistent with a scaling of the aluminium confinement time $\tau_{Al} \propto a_p^{2.4} \cdot \hat{n}_e^{1.2} \cdot B^{0.3} / P_{ECRH}^{0.8}$ (plasma radius a_p , central electron density $\hat{n}_e < 5 \cdot 10^{19} \text{m}^{-3}$,

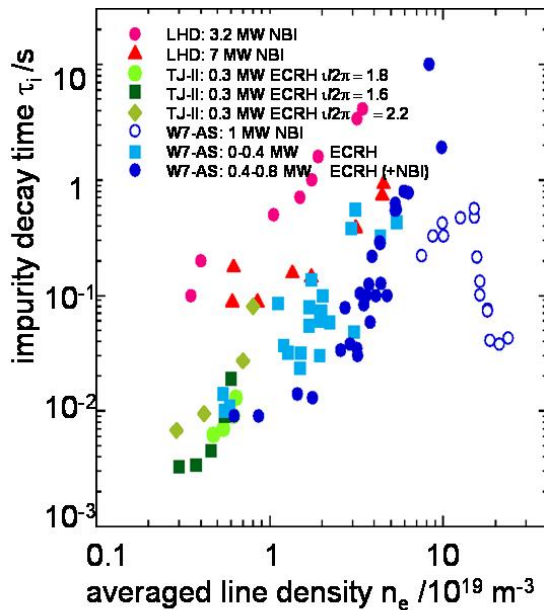


FIG. 1. Impurity confinement time vs. density in TJ-II, LHD, W7-AS

magnetic field B , electron cyclotron radiation heating (ECRH) power P_{ECRH}) and a density and density gradient dependent inwards convection velocity v in LHD [7]. In LHD, which has a large flexibility in changing the ambipolar radial electric field E_r by controlling the effective ripple and the magnetic topology, the impact of E_r on impurity transport was studied [6,8]. Consistently, a radial electric field evolving gradually from positive (outward flow of impurities) to negative (inward flow) values with growing density was found. In addition, a reduction of turbulent transport in both machines at higher density is assumed to account for the density scaling of the impurity behaviour. Although direct evidence of turbulent transport is difficult, there are some convincing indications pointing to the essential role of anomalous impurity transport,

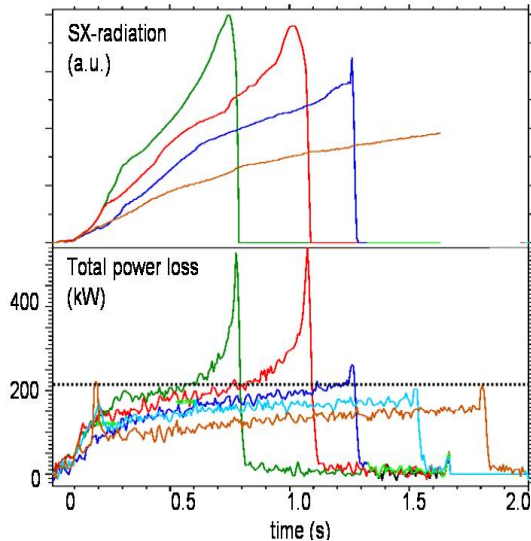


FIG. 2. Impurity radiation in 0.5 MW NBI-heated W7-AS plasmas at constant density of 0.87 (brown and light blue, #42721/42714) / 1.05 (dark blue, #42709) / 1.15 (red, #42706) / $1.25 \cdot 10^{19} \text{ m}^{-3}$ (green, #42698). The dashed line indicates 40% of heating power [9].

in particular at lower and medium density [4,5]: 1. The impurity diffusion coefficient D decreases with density (W7-AS) or is independent of density (LHD). This is in contrast to conventional neoclassical scaling, which predicts an increase of D with density for both cases. Transport simulations at LHD [5,7] reveal diffusion coefficients being one order of magnitude enhanced over the

neoclassical level for low and medium density. Towards high densities experimental and neoclassical D 's seem to approach each other. 2. At W7-AS both impurity and energy confinement obey quite similar dependences on global parameters (density, heating power etc.) suggesting common driving mechanisms.

The consequences of the unfavourable impurity confinement at high density manifest themselves in radiation losses that steadily rise throughout the pulse duration by accumulating impurity ions [4,6]. At fractions of radiation power losses larger than 40-50% of the heating power the plasma energy starts to degrade in W7-AS and the discharge terminates by radiation instability (FIG. 2.). The maximum achievable pulse length becomes therefore a function of the electron density for a given heating power and impurity concentration. The high MHD stability limits do not restrict the operational range in these discharges and are not responsible for the energy degradation. Hence, the radiation losses determine the density limit in W7-AS, which – in contrast to tokamaks - is a soft limit and depends on heating power and impurity concentration. Therefore, both, degradation of impurity confinement in the bulk plasma as well as screening of impurity influx at the edge are of equal necessity for successful long-pulse operation.

3. General Approaches for Reduction of Impurity Radiation Losses

3.1. Reduction of Impurity Sources

Generally, impurity accumulation does not necessarily lead to a radiation collapse if the influx is kept small: for densities $n_e > 5 \times 10^{19} \text{ m}^{-3}$ successive onset of accumulation is observed in W7-AS limiter ECRH discharges due to an overall reduction of D and increase of v_{inward}/D in particular at the plasma boundary, where the usually peaked profiles of D have their smallest values. Central fluorine radiation continued rising while the fluorine influx was kept constant by an external gas valve in W7-AS. This behaviour could be well simulated by means of a reduced D at higher density, which is assumed to be a consequence of reduced turbulent transport. However, the simulations predict a saturation of the power loss after a time constant of 1.7s (longer than the technically limited pulse length of 1s) which is determined by the edge diffusion coefficient. Extrapolation to 1.7s reveals a saturation level for the radiation loss of 40% of the heating power in this particular case, which is marginally below the level where degradation of the plasma energy by total radiation losses is expected. Because the final saturation level is directly related to the impurity source strength, an accumulating plasma can

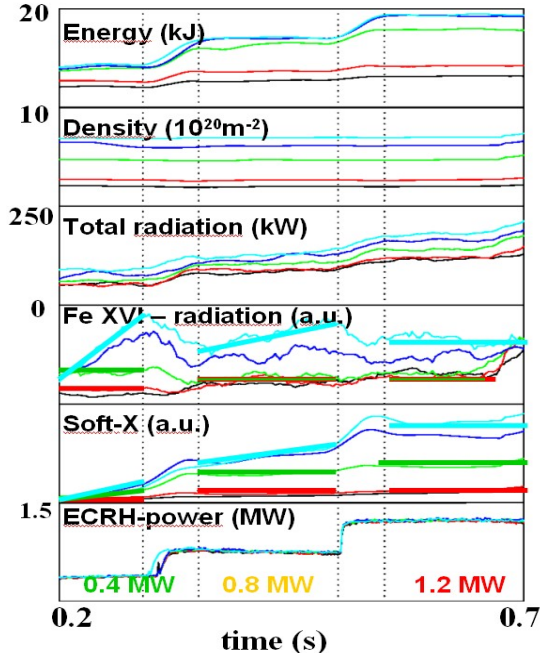


FIG. 3. Evolution of impurity radiation in plasmas with different density and heating power (#43133, 43134, 43140, 43143, 43144).

principally be maintained stationary if the sources can be kept sufficiently small. This can be achieved by carefully shielding the plasma facing in-vessel components as a first “passive” approach (more sophisticated scenarios see sec. 4). The radiation losses are mainly dominated by VUV radiation from low/medium ionization states and therefore inversely scaling with the electron temperature. Sufficient heating power can avoid early limitation of the pulse length.

3.2. Utilization of Power Degradation of Impurity Confinement

Utilization of the power dependence of the impurity confinement time (see confinement time scaling in section 2 and power degradation of impurity confinement in FIG. 1.) is a pragmatic tool to exert influence on the impurity concentration. The degradation of confinement at higher ECRH power is a consequence of enhanced D in W7-AS and both, enhanced D and changing ν in TJ-II, which might indicate an increased anomalous transport. As demonstrated in W7-AS-discharges with three consecutive ECRH heating power steps performed at five different densities, accumulation can be successfully suppressed by application of sufficient heating power, e.g. 1.2 MW at a central density of $9 \cdot 10^{19} \text{ m}^{-3}$ (FIG. 3.). This corresponds to a volume-averaged power density of approximately 1 MW/m^3 . The existing restriction of ECRH heating to densities below the cut-off density ($1.2 \cdot 10^{20} \text{ m}^{-3}$ for 140GHz ECRH second harmonic X-mode resonant absorption) can possibly be overcome by new microwave heating scenarios such as the Electron-Bernstein Wave heating (EBW). The Electron-Bernstein waves are excited via the O-X-B mode conversion process and extends the operational density range with no upper limit. This heating scenario might pave the way for high density ECRH operation and the utilization of power degradation even at high density [10].

4. Particular Purification Features in High Performance Plasmas

Numerous improved confinement modes at medium and high density were observed in helical devices, such as e.g. High T_i (-like) mode at HELIOTRON E and CHS, edge transport barrier (ETB) and Reheat-mode (improved L-mode) plasmas at CHS, H-mode like transitions and Super Dense Core plasmas in LHD, and optimum confinement (OC, earlier:H-NBI) as well as H/HDH-mode plasmas in W7-AS. Some of them are affected by impurity accumulation phenomena which often prevent them from stationary performance – some of them don’t. Nevertheless, the comparison of these scenarios might disclose potential ways out of the dilemma and are discussed below.

4.1. Impurity Influx Shielding in the Edge/SOL of Divertor Plasmas

Both, the helical divertor in LHD and the island divertor in W7-AS (and W7-X) demonstrate favourable impact on the impurity retention [11,12] in the SOL. Generally in both, tokamaks and non-axisymmetric devices, the parallel ion temperature gradient along the separatrix

(“upstream”) to the lower temperature in front of the divertor plates (“downstream”) causes a competition between downstream directed frictional plasma flow flushing the impurity ions back to the divertor plates and parallel upstream directed impurity flux induced by thermal forces on the impurity ions. The latter depends on the temperature gradient ($\propto \nabla T_i$) and, via the collision frequency, quite sensitively on T_i ($\propto T_i^{3/2}$). A reduction of the impurity influx can be achieved by reduction of ∇T_i or T_i . When the parallel heat conduction governs the energy transport in the SOL like in tokamaks, there is no direct way to control ∇T_i and T_i independently. A reduction of T_i will cause steepening of the temperature gradient, yielding no net effect of reducing the impurity influx. Non-axisymmetric divertors based on islands, however, have the potential to decrease ∇T_i and T_i simultaneously by additional cross-field heat transport through the islands which grows with density and acts like a supplementary bypass between up- and downstream positions. This causes a strong reduction of the parallel conductive heat flux ($\propto T_i^{5/2}$) and finally of the thermal impurity influx. As a consequence, the impurity density at the separatrix as well as the impurity influx into the core region [11] is decreased. In detail, this might differ depending on the real geometry of the divertor structure.

This impurity screening mechanism was well simulated by the EMC3/EIRENE code and experimentally demonstrated in the SOL of W7-AS during the HDH-mode [11] and also in the edge ergodic and surface layer of LHD [12]. With the onset of the HDH mode in W7-AS at central densities of $1.5\text{-}2.2 \cdot 10^{20} \text{ m}^{-3}$ ($P_{\text{NBI}} = 1\text{-}3.5\text{MW}$) and edge densities above $1\text{-}2 \cdot 10^{19} \text{ m}^{-3}$ the impurity influx remarkably reduces and enables stationary operation at high density up to $4 \cdot 10^{20} \text{ m}^{-3}$ without accumulation - which was not possible before. An essential ingredient for achieving high edge density (and low edge temperature) was the use of strong gas puffing. This transiently forces the formation of a broad H-mode like density profile with steep edge gradients. Large negative radial electric fields E_r/T_i at the plasma boundary, suppressed fluctuation activity and the resulting reduced edge transport are stabilizing the HDH-mode self-consistently. In LHD, the trend for accumulation in long pulse experiments ($P_{\text{NBI}} = 1.5\text{MW}$) at medium density (accumulation window $1.0\text{-}2.7 \cdot 10^{19} \text{ m}^{-3}$) was stopped beyond a relatively low (compared to W7-AS) power dependent line averaged density of $2.7 \cdot 10^{19} \text{ m}^{-3}$. However, the impurity core confinement time (impurity pellet injection experiments) still remains long (!) beyond the accumulation window, indicating that the onset of purification is not exclusively determined by the core confinement: the injection experiments are sensitive to the core confinement whereas the global intrinsic impurity behaviour might be dominated by edge shielding processes, which is no contradiction. Likewise, the Super Dense Core (SDC) plasmas with internal diffusion barrier ($n_e(0)=5\text{-}6 \cdot 10^{20} \text{ m}^{-3}$, $T_e(0)=660\text{-}850 \text{ eV}$) obtained in LHD by frequent pellet injection in the presence of a helical divertor or the local island divertor (LID) do obviously not suffer from enhanced radiation, although impurity accumulation and long impurity confinement was observed [13,14,15]. In both cases a very effective shielding of the impurity influx in LHD is suggested which can most likely be assigned to the same intrinsic edge screening as discussed above, so that the observations at LHD and W7-AS seem to be consistently explainable by the same screening mechanism.

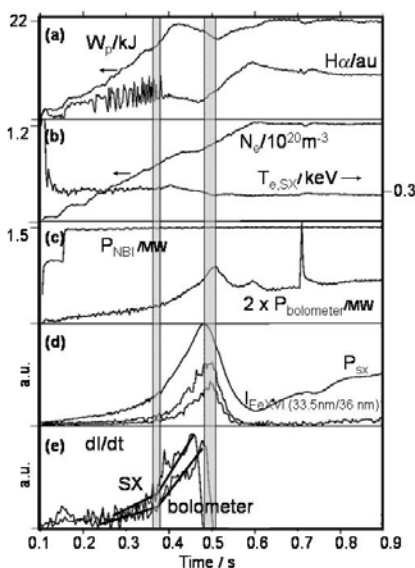
In the long pulse experiments at LHD even with densities inside the accumulation window impurity control is also successively demonstrated during generation of a $n/m = 1/1$ magnetic island near the plasma edge by means of external perturbation coils. After application of the perturbation field the initially rising central radiation decreases remarkably on a large time scale, the radiation profile centrally dropping from peaked to hollow. This may be interpreted as the consequence of a significantly reduced impurity influx [6], the exact mechanism still being under discussion. During additional operation of the LID in the magnetic island

configuration a positive radial electric field with a strong shear was produced at the plasma edge by electron loss to the LID, triggering an internal transport barrier (ITB) and reducing the impurity influx [8]. All these phenomena stress the relevance of impurity edge transport for successful prevention of excessive radiation losses and stationary performance - in addition to the core confinement.

4.2. Degradation of Impurity Core Confinement at High Density

Simultaneously with the onset of the screening of impurity influx into the W7-AS HDH plasma, also the impurity core confinement drops dramatically at high density (see FIG. 1.) in contrast to LHD. With the onset of the HDH mode the density profile suddenly flattens whereas the temperature profiles remain peaked. Both improved energy and degraded impurity core confinement (measured by impurity injection technique) could be demonstrated [4], which is favorable for fusion devices but difficult to achieve simultaneously. At these high densities impurity transport simulations for HDH reveal nearly neoclassical character. One of the reasons for the sudden drop of impurity confinement is a reduced inward core convection which is neoclassically consistent with the flattening of the density profiles. However, neoclassical simulations additionally predict strong inwards directed impurity fluxes being inferred by the strong edge density gradients. This was experimentally substantiated by derivation of a narrow layer of strong negative radial electric field E_r/T_i from the edge pressure profile, driving the inwards flux. The existence of this layer is predicted to cause long impurity confinement times in contrast to the observations. Finally, the introduction of an artificially enhanced impurity edge diffusion coefficient, representing some still unknown transport mechanisms locally counteracting the impurity inwards convection, was necessary to match the extremely small observed impurity confinement times (down to 10ms). Code calculations show that changes of this anomalous edge diffusion coefficient and hence changes in the two balancing edge transport processes have substantial impact on the global impurity confinement – ranging from extremely short to nearly infinite impurity confinement times. This additionally underlines the importance of edge/SOL transport for the global impurity confinement. However, the achievement of HDH properties also in high- β limiter plasmas indicates that a divertor configuration might not necessarily be a prerequisite.

4.3. Impact of Edge Localized Modes (ELMs)



At W7-AS, two improved edge confinement modes exist at medium density (approx. $8 \cdot 10^{19} \text{ m}^{-3}$) which do not benefit from impurity screening inferred by high SOL collisionality as described in section 4.1 and have to utilize other mechanisms: 1. The normal H-mode has flat core density profiles and steep edge gradients similar to HDH, but not as high edge density and low edge temperature as needed for effective friction induced edge screening. 2. The standard ion root

FIG. 4. W7-AS discharge #56156 with transition from H-mode to H*-mode (around 0.37s) and then to HDH-mode (around 0.5s). (a) Plasma energy W_p , H_α , (b) average density N_e , electron temperature derived from SX-radiation $T_{e,SX}$, (c) NBI and total radiation power, (d) SX-radiation power and FeXVI line radiation, (e) derivative of bolometer and SX radiation power.

optimum confinement regime (OC, similar to High T_i modes in CHS and HELIOTRON E [16]) with peaked density profiles, inward shifted density gradients (often larger than in ELM free H-modes (H^*)), low recycling and broad ion temperature profiles. In the normal H-mode the already long impurity confinement according to the density scaling is even enhanced, and the absence of ELMs (ELM free H-mode H^*) leads to an exponential increase of radiation provoking an abrupt termination of the pulse. If the density is ramped up quickly enough to more than $1.5 \cdot 10^{20} \text{ m}^{-3}$, again stationary operation without accumulation in the HDH-mode becomes possible (FIG. 4.). Nevertheless, the residence time in the H^* -mode has to be kept short in order to avoid enrichment of the core plasma by impurity ions and thus the onset of energy degradation by enhanced radiation before reaching the HDH regime. The NBI-heated OC-regime is also afflicted by accumulation but cannot be treated by rigid procedures such as strong impurity puffing as discussed in section 4.1. The OC-regime is the result of a complex bifurcation process evolving gradually on a particle transport time scale under defined conditions for NBI heating power, density, recycling and additional ECRH-power for density control. The highest ion temperatures and longest energy confinement times at W7-AS were achieved in this regime, which was found to behave almost neoclassically.

In both modes ELMs develop with a large variation in their frequency in the H-mode and often a grassy character in OC plasmas, which might be utilized to prevent accumulation. The beneficial impact of ELMs on impurity concentration in H-modes but also their drawbacks for fusion reactors with respect to excessive power loads onto plasma facing in-vessel components are permanent subjects of investigations at tokamaks. During the last years, at TJ-II, LHD and W7-AS extensive studies on the nature of ELMs or ELM-like bursts in Stellarators/Heliotrons [17,18,19] were performed. With respect to the general trend for improved core impurity confinement at high density, however, it is worthwhile to elucidate also the role of ELMs as possible candidates for plasma purification in helical systems. FIG. 5. shows the onset of increasing core radiation (central channel of Soft-X-camera) starting with the formation of the H^* -mode and a flattening during the onset of frequent ELM activity. The time trace of the Al XII emission after a short aluminium laser blow-off injection reveals nearly no significant decay even in the phase with ELMs, indicating still long impurity confinement in the presence of saturating bulk radiation. Consistent to the experiences from tokamaks, these are indications that frequent ELMs have impact on the net impurity influx into the core plasma and can principally be helpful in the suppression of accumulation in both

high performance scenarios. Large single ELMs have significant impact on the edge radiation profiles but reveal less effective impurity shielding than frequent ELM activity. The challenging task is, of course, to identify windows in the configurational space to control the appearance of ELMs.

4.5. Summary

Stellarators/Heliotrons have the intrinsic potential for continuous operation and focus on the achievement of

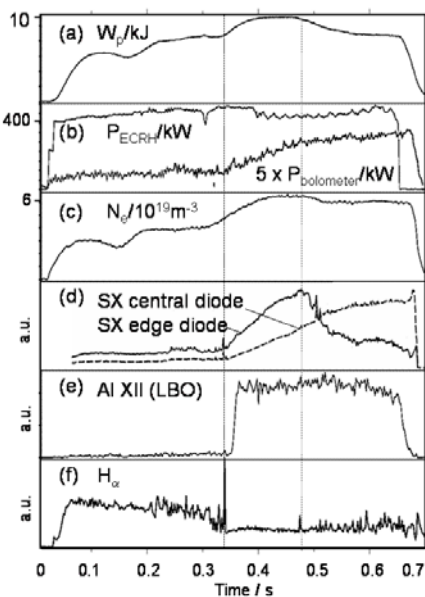


FIG. 5. W7-AS discharge #21142 with L-H transition at 0.34s. (a) Plasma energy W_p , (b) ECRH heating and total radiation power, (c) average density, (d) radiation of central and edge (inside ELM pivot point) SX-diode, (e) Al XII radiation after Al laser blow-off) at 0.35s, (f) H_α

high performance plasmas which can be maintained stationary. However, theoretical predictions basically forecast accumulative impurity behaviour in non-axisymmetric devices in the standard ion root regime. Moreover, experimental findings reveal some unfavorable density dependence of the impurity confinement when going to dense or improved confinement plasmas. Nevertheless, various operational scenarios were already explored in present generation Stellarators/Heliotrons which provide pragmatic approaches on one hand (reduction of impurity sources, heating power degradation of impurity confinement) or favourable edge transport mechanisms on the other hand (frequent ELMs, impurity screening by friction force, introduction of $n/m = 1/1$ magnetic islands and positive radial electric fields at the plasma edge) to effectively control impurity influx/confinement. The latter are specific advantages of non-axisymmetric devices and can partly be quite well understood from classical theory. Additionally, degradation of impurity core confinement was obtained in HDH, the global confinement obviously depending also sensitively on transport processes at the plasma edge. This stresses the importance and necessity to include three-dimensional effects additionally to the conventionally used axisymmetric approximations of neoclassics in impurity transport modelling - not only to understand current observations but also to explore further operation regimes. Both the role of turbulent transport with respect to power degradation of impurity confinement and the edge transport processes in HDH still need continuative investigations for better understanding. The effect of ELMs on the reduction of the net impurity influx was shown but their controlled applicability needs further exploration. From the inter-machine comparison it can be finally concluded, that - in spite of the primary unfavourable density scaling of the global impurity confinement - effective screening of impurity seems possible in Stellarators/Heliotrons providing stationary radiation power loss conditions on an acceptable level.

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