Overview of First Wendelstein 7-X High-Performance Operation with Island Divertor

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The optimized superconducting stellarator device Wendelstein 7-X restarted operation after the assembly of a graphite heat shield and an inertially cooled island divertor. This paper reports on results from the first high-performance plasma operation. Plasma densities of $1\text{–}4 \times 10^{19}/\text{m}^3$ with electron temperature $5\text{–}10$ keV were routinely achieved with hydrogen gas fuelling, eventually terminated by a radiative collapse. Up to $1.4 \times 10^{20}/\text{m}^3$ plasma density was reached with repetitive hydrogen pellet injection. Here, the ions are indirectly heated, and at a density of $8 \times 10^{19}/\text{m}^3$ temperatures $T_e \approx T_i = 3.4$ keV were accomplished, which corresponds to $nT\tau_E = 6.4 \times 10^{19}$ keV s/\text{m}^3 with peak diamagnetic energy 1.1 MJ. Stable 25 s long-pulse helium discharges with 2–3 MW ECRH power and up to 75 MJ injected energy were created routinely for equilibrium and divertor load studies, with plasma densities around $5 \times 10^{19}/\text{m}^3$ and 5 keV electron temperature. The divertor heat loads remained far below the limits. The O/C impurity concentration ratio has decreased in comparison to the previous limiter operation and no intrinsic impurity accumulation along with high edge radiation were observed in stationary plasmas. During pellet-fuelled hydrogen discharges, full detachment was observed with divertor target heat flux reduction by more than $\times 10$. Both X2 and O2 mode ECRH schemes were applied and electron cyclotron current drive (ECCD) experiments were conducted. During co-ECCD injection experiments with axial currents up to 13 kA, frequent fast crashes were observed mainly in the core electron temperature, suggesting a fast magnetic reconnection mechanism. The radial electric field measured with (Doppler) and correlation reflectometry changes sign at the plasma edge from $+10 \ldots +20$ kV/m to $-10 \ldots -5$ kV/m, fairly independent of discharge parameters and heating power. Edge and scrape-off layer turbulence was measured with both Langmuir probes and reflectometer diagnostics. Core turbulence was measured with a phase contrast imaging diagnostic and different levels of broad band turbulence as well as coherent Alfvén mode activity were observed.
OVERVIEW OF FIRST WENDELSTEIN 7-X HIGH-PERFORMANCE OPERATION WITH ISLAND DIVERTOR

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Abstract

The optimized superconducting stellarator device Wendelstein 7-X (with major radius \( R = 5.5 \) m, minor radius \( a = 0.5 \) m, and 30 m\(^3\) plasma volume) restarted operation after the assembly of a graphite heat shield and 10 inertially cooled island divertor modules. This paper reports on the results from the first high-performance plasma operation. Glow discharge conditioning and ECRH conditioning discharges in helium turned out to be important for density and edge radiation control. Plasma densities of \( 1.4 \times 10^{19} \) m\(^{-3}\) with central electron temperatures 5-10 keV were routinely achieved with hydrogen gas fuelling, eventually terminated by a radiative collapse. Plasma densities up to \( 1.4 \times 10^{20} \) m\(^{-3}\) were reached with hydrogen pellet injection and helium gas fuelling. Here, the ions are indirectly heated, and at a central density of \( 8 \times 10^{19} \) m\(^{-3}\) a temperature of 3.4 keV with \( T_e/T_i = 1 \) was accomplished, which corresponds to \( n_e T_e(0) \tau = 6.4 \times 10^{19} \) keVs/m\(^3\) with a peak diamagnetic energy of 1.1 MJ. Stable 25 s long-pulse helium discharges with 2-3 MW ECRH power and up to 75 MJ injected energy were created routinely for equilibrium and divertor load studies, with plasma densities around \( 5 \times 10^{19} \) m\(^{-3}\) and 5 keV electron temperature. The divertor heat loads remained below the limits, even at highest heating powers. The O/C impurity concentration ratio has noticeably decreased in comparison to the previous limiter operation. No intrinsic impurity accumulation and high edge radiation were observed in stationary plasmas with densities up to \( 4 \times 10^{19} \) m\(^{-3}\). In pellet-fuelled hydrogen discharges, full detachment was observed with divertor target heat flux reduction by more than \( \times 10 \). The discharge behaviour has further improved with boronization of the wall. After boronization, the oxygen impurity content was reduced by a least \( \times 5 \), the carbon impurity content by at least \( \times 2 \). The reduced (edge) plasma radiation level gives routinely access to higher densities without radiation collapse, e.g. well above \( 1 \times 10^{20} \) m\(^{-2}\) line integrated density and \( T_e/T_i = 2 \) keV central temperatures at moderate ECRH power. Both X2 and O2 mode ECRH schemes were successfully applied and electron cyclotron current drive (ECCD) experiments for divertor target strike line control were conducted. During co-ECCD injection experiments with broadly distributed total toroidal currents of up to 18 kA, fast plasma crashes were observed, suggesting a fast magnetic reconnection mechanism potentially associated with low-order resonant \( i/2\pi = 5/5 \) values. Edge and scrape-off layer turbulence was measured with both Langmuir probes and reflectometer diagnostics. Core turbulence was measured with a phase contrast imaging diagnostic and different levels of broad band turbulence were observed.

1. INTRODUCTION

After successful first operation [1] in 2015, the optimized stellarator device Wendelstein 7-X [2, 3] is now running with graphite heat shields and a graphite island divertor [4, 5]. Wendelstein 7-X is a high-iota low shear stellarator with optimized magnetic field and 30 m\(^3\) plasma volume. It is the mission of the device to demonstrate steady-state creation (1800 s) of fusion-relevant hydrogen and deuterium plasmas [3]. The magnetic field with 2.5 T on the magnetic axis is generated with a set of 50 non-planar and 20 planar superconducting NbTi coils. In addition, all plasma facing components are designed for active water cooling. Steady-state plasma heating is provided by 10 gyrotrons exclusively. The operation phase reported in the present paper is yet without water cooling of the in-vessel components. This restricts the heating energy input to about 80 MJ, which nevertheless allows for high-performance plasma operation but at limited pulse lengths (typical 5-50 s). At these scenarios, actual divertor operation must be demonstrated to develop the basis for high-performance steady-state operation in the next operation phase that follows after completion of the cooling water systems and the installation of the water-cooled divertor and the cryo pumps. The paper is structured in a brief machine description, three main physics sections, and a conclusion: The first physics section is about high-performance plasmas, the second one about the verification of stellarator optimization, the third section about the operation of the island divertor.
2. THE WENDELSTEIN 7-X STELLARATOR DEVICE

The magnetic field of the superconducting stellarator Wendelstein 7-X was optimized to address main issues of classical stellarator, namely good MHD equilibrium and stability properties, strongly reduced neoclassical transport, reduced bootstrap current (for controlled edge iota profiles), improved confinement of fast particles [6].

![Schematic diagram of the Wendelstein 7-X device. Shown are the non-planar and planar coils (red resp. orange), the bus bar and helium supply, the central support ring, one module of the outer vessel (grey), and the last-closed magnetic flux surface (blue).](image)

**Figure 1** Left: Schematic diagram of the Wendelstein 7-X device. Shown are the non-planar and planar coils (red resp. orange), the bus bar and helium supply, the central support ring, one module of the outer vessel (grey), and the last-closed magnetic flux surface (blue). Right: Radial profiles of the rotational transform \( \iota \).

A schematic diagram of the device is shown in Fig. 1. The 50 non-planar coils (red) and 20 planar coils (orange) are connected in series via superconducting bus bars. All coils are bolted to a massive central support ring (grey) and additionally fixed by mostly welded, partially bolted or sliding local support elements. The main device parameters are listed in Tab. 1. Stage I was the setup for the initial operation. After substantial extension in the in-vessel components and the heating systems, the present stage II was reached. Stage III is planned for the subsequent operation phase. Stage IV is the projected full performance configuration of the device.

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**Table 1** Main device parameters of Wendelstein 7-X. The different stages refer to completion steps, mainly of the heating systems and the water cooling of the in-vessel components. The present status is stage 2.

The main heating scheme is electron cyclotron resonance heating (ECRH) with at present 10 long-pulse capable 140 GHz gyrotrons [6]. On average each gyrotron accounts for 0.8 MW power coupled into the plasma which provides a highly flexible X2-mode and O2-mode heating scheme, both on- and off-axis. Both the flexibility and the well-defined heat deposition in the electron cyclotron resonance zone render ECRH being the most important heating and current drive scheme with the biggest potential for a future stellarator power reactor [6]. The first of two neutral beam injector (NBI) boxes has started operation with two positive ion sources and 55 kV acceleration voltage and up to 3.5 MW injection power [7]. The ion cyclotron resonance heating (ICRH) system requires an antenna that is carefully shaped to the three-dimensional plasma contour. This development is ongoing and commissioning is foreseen for the next operation phase [8]. To protect the many uncooled in-vessel components, the maximum heating energy during stage 2 is at present limited to 80 MJ, which implies typical discharge times between 5 and 50 s, depending on the input power. After completion of the water cooling systems and the replacement of the inertially cooled island divertor with an actively water cooled one, the maximum heating energy will be extended to 18 GJ with at least one intermediate step at 1 GJ (stage 3). Wendelstein 7-X started
first operation in 2015. The heat and particle exhaust was controlled with five poloidal graphite limiters, the remaining wall was either steel or CuCrZr. Despite the unfavorable wall conditions, the plasma performance was quite remarkable with peak electron temperatures ≥8 keV with simultaneous peak ion temperature ≥2 keV and line averaged density 3·10^{19} m^{-3} [1]. These are typical conditions for the core electron root confinement [9] which is characterized by a reversal of the radial electric field from edge to core [10]. First elements of stellarator optimization could be demonstrated by studying the bootstrap current and neoclassical transport [11].

The aim of the second operation phase (at stage 2) is to demonstrate divertor operation with improved plasma performance and exhaust after the limiters have been replaced by an island divertor and major parts of the wall are covered with graphite tiles. The island divertor consists of ten separate modules formed by graphite target and baffle plates that are matched to the magnetic field structure of Wendelstein 7-X [4, 5]. Depending on the magnetic configuration (especially the edge rotational transform τ/2π, cf. Fig 1.) natural magnetic islands form at the plasma boundary. They are intersected with the divertor target plates and thus establish a multi X-point divertor for the exhaust of particle and heat flows via the last closed flux surface.

3. HIGH PERFORMANCE DISCHARGES

![Figure 2](image-url) **Figure 2** Plasma discharges (a) before and (b) after boronization. Shown are (from top to bottom) the ECRH power and the radiation power, the line-integrated density and the hydrogen gas inlet, the central electron (ECE) and ion temperatures (from XICS), and the diamagnetic energy.

The plasma performance of Wendelstein 7-X has dramatically improved after the installation of the graphite heat shields and the graphite island divertor. Another significant step forward was made with appropriate wall conditioning [12, 13]: The plasma vessel is baked at 150 °C after venting in order to remove water and hydrocarbons from the wall and in-vessel components. Without magnetic field, glow discharge cleaning (GDC) is applied in hydrogen (to reduce residual CO and CH_4) and helium gas (to reduce H_2). With the superconducting magnets ramped up, an additional wall conditioning between discharges was made with ECRH short pulse trains followed by pumping intervals. Boronization was conducted for the first time in the beginning of the second half of the (stage 2) experimental campaign. Here, a glow discharge with diborane (B_2H_6) in helium background gas was operated for 4-5 hours. The GDC in helium and hydrogen as well as the occasional ECRH pulse train conditioning of the plasma facing components (with total surface areas of 110 m^2 graphite and 80 m^2 steel) have greatly reduced the outgassing rates, rapidly dropping to values that were reached only at the end of the previous (initial stage 1) operation phase with graphite limiters. Before boronization, plasma densities of 1·10^{19} m^{-3} with electron temperature 5-10 keV were routinely achieved with hydrogen gas fuelling, where higher densities where inaccessible due to radiative collapse. Higher densities where not accessible due to radiative limits (see also Fig. 4 below). However, a peak plasma density up to 1.4·10^{20} m^{-3} could be reached with repetitive hydrogen pellet injection and simultaneous ECRH with the second harmonic O-mode (O2-heating) [14]. In this density range the ions are indirectly heated, and at a central density of 8·10^{19} m^{-3} a temperature of 3.4 keV with T_e/T_i=1 was accomplished with X2-heating, which corresponds to nT_e T_i=6.4·10^{19} keVs/m^3 with a peak diamagnetic energy of 1.1 MJ [15]. Stable 25 s long-pulse helium discharges with 2-3 MW ECRH power and up to 75 MJ injected energy were created routinely for equilibrium and divertor load studies, with plasma densities around 5·10^{19} m^{-3} and 5 keV electron temperature. The first application of a glow discharge with diborane deposited a boron layer
of about 100 nm thickness which is lasted for about 200 plasma seconds at the strike line position of the divertor target and for about 2000 plasma seconds on other wall elements [13]. Fig. 2 shows the difference between the discharge conditions (a) before and (b) after boronization. With the same ECR heating power and standard gas puffing, about ×3 higher plasma densities could be reached without radiative collapse. This improvement is owing to the reduced O and C content, which strongly reduces the radiative power losses predominantly in the plasma edge, where the radiative collapse usually starts. At higher densities ~1·10^{20} \text{m}^{-3} electrons and ions equilibrate and T_e=T_i=2 \text{ keV} is reached. Correspondingly, the diamagnetic energy is higher than before boronization (at other plasma parameters). A spectroscopic measurement of OV and OVI as well as CV and CVI ions before and after boronization is shown in Fig. 3. The spectroscopic signals are normalized to the line-averaged plasma density measured for the respective discharge. Note that for the lower wave lengths, no absolute calibration is available and temperature and profile effects have not yet been taken into account. Nevertheless, a strong reduction of the spectral line strength by ÷2.5 for carbon and ÷6.5 – 8 for oxygen is evident and a similar reduction in the associated impurity concentration can be assumed. This gives strong evidence for the expected gettering effect by the boron layer on the plasma facing components [16].

![Figure 3](image1.png)

**Figure 3** Spectral lines of OV and OVI, CV and CVI ions before (blue) and after boronization (green). The BV is also seen after boronization. Spectral line strengths of the C and O ions are significantly reduced after boronization. From Ref. [28].

An immediate consequence of the reduced impurity concentration is a strong shift in the radiative density limit, as shown in Fig 4(a). Before boronization, the critical plasma density \( n_c \) for radiative collapse at increasing heating power remains somewhat below the empirical power scaling discussed in Ref. [17] for an impurity concentration fraction \( f_{\text{imp}}=n_{\text{imp}}/n_e =5\% \). In contrast, after boronization \( n_c \) shifts in the same heating power range to about ×3 higher values and shows a more square root like power dependence and follows the analytic scaling law for impurity concentration fractions between 0.5-1%. In order to assess the impact of the plasma radiation close to the density limit on \( \tau_E \), the scaling factors (\( \alpha, \beta \)) of the energy confinement time for density \( n_e \) and heating power \( P \) as a function of the radiative power \( P_{\text{rad}} \) normalized to the ECR heating power \( P_{\text{ECRH}} \) are shown in Fig. 4 (b).

![Figure 4](image2.png)

**Figure 4** (a) Radiative density limit before (green symbols) and after boronization (blue symbols). After boronization, the critical density increases by a factor of ~3 and shows a stronger, square-root-like power dependence. For comparison an analytic scaling law is included in the diagram for three different impurity concentration fractions 0.5 %, 1% and 5%. (b) Scaling factors of the energy confinement time for density (left) and heating power (right) for different radiation fractions. From Ref. [17].

At high radiation fractions > 50%, a clear tendency to a weaker density scaling and power degradation due to radiative losses is seen. A degradation of the stored energy with increasing density, however, only occurs at densities close to a radiative collapse. For a more comprehensive discussion on this topic, the reader is referred to
Ref. [17] at this conference. With the benefits of boronization, stationary high density operation with high power O2 ECRH above the X-mode cutoff density of 1.2·10^{20} m^{-3} became accessible even with simple hydrogen gas fuelling. Both high performance plasmas and low divertor loads have been achieved for 12 s, limited only by the maximum allowed input energy of 80 MJ. More details are presented in Ref. [14] at this conference.

4. STELLARATOR OPTIMIZATION

It is the ultimate goal of Wendelstein 7-X to demonstrate the beneficial effect of the optimized magnetic field geometry, in particular reduced neoclassical transport and bootstrap current along with magnetic islands well-localized at the edge and good magneto-hydrodynamic stability. Fast particle confinement is another aspect of optimization. The sum of all should result in much improved plasma performance and eventually to fully integrated, stable plasma scenarios with simultaneously high \( T_e - T_i \) at high particle densities and energy confinement times \( \tau \) better than the ISS04 scaling [18]. Stable high-performance plasma scenarios require a sufficient control of the plasma-wall interaction (in particular the impurity source), the heat and particle exhaust as well as the particle recycling. Details are discussed in the subsequent section 5. Below we present a number of experimental results that give further evidence for the success of stellarator optimization. Fig. 5(a) shows the Rogowski coil measurement of the time evolution of the net toroidal plasma current for plasmas with constant ECR heating power in different magnetic field configurations: low iota (magenta line, label DAM), standard (blue lines, labels EIM and EJM), high mirror (green lines, label KKM), high iota (black lines, label FTM). The dashed red lines are exponential fits of the measurements. A prediction based on one-dimensional transport modelling is shown in Fig. 5(b). The transport code requires as input the measured \( n_e \) and \( T_e \)-profiles (not available for DAM). The time scale, mainly given by the \( R/L \)-time \( \sim 10 \) s, and the expected strong reduction of the bootstrap current in high-mirror and high-iota configurations is well confirmed by both measurement and simulation, except for a systematic shift by 2-3 kA current. This shift might well be due to an (unintended) electron cyclotron current drive component which is currently under investigation. First findings on the reduction of the bootstrap current owing to magnetic field optimization were already reported from the previous operation campaign without divertor [11].

![Figure 5](image)

**Figure 5** (a) Time evolution of the toroidal current measured with Rogowski coils. The different colors indicate the different magnetic configurations (see text). The dashed lines are exponential best-fits. (b) One-dimensional transport code simulations of the toroidal current based on experimental density and temperature profiles for the same magnetic configurations. After Ref. [29].

The long discharge duration allows one to make use of electron cyclotron current drive (ECCD) for a feed-forward control of the edge rotational transform [14], which is changed by the toroidal current evolution that strongly depends on the magnetic configuration and the discharge parameters. The compensation of the toroidal current by counter-ECCD was successfully demonstrated and the strike line position on the divertor targets [19, 20] could be well controlled [14] (see Sec. 5 below). For a finite positive toroidal net current, the rotational transform \( \iota \) is increased and the island chain is radially shifted inward towards the magnetic axis which in turn displaces the strike line on the target away from the divertor pumping gap. Central co-/counter ECCD leads under certain discharge conditions to fast repetitive electron temperature collapses in the core or even a total plasma collapse, most likely due to crossing the resonant \( \iota/2\pi=1 \). Further details on the successful application of ECR X2/O2
$T_e=6$ keV and $T_i=2$ keV. The density profile is flat up to $r_{95a} \approx 0.8$ ($\pi r_{95a}$ is the toroidally averaged poloidal cross sectional area) and the temperature profiles are peaked in the center. The radial diffusion and convection profiles were obtained by fitting the emission line time traces of various ionization stages (see inset in Fig. 7) using combined neo-classical and anomalous $D_n+D_v$ and $v_n+v_v$ profiles in the STRAHL code. It turns out that the anomalous contributions are two orders of magnitude above the neo-classical level [24]. The impurity transport is clearly dominated by turbulence, either trapped-electron mode (TEM) or ion temperature gradient (ITG) driven turbulence (or both) in the radial pressure gradient regions around $r_{95a} \approx 0.8$. Impurity accumulation is considered as a major stellarator issue, since especially in the ion-root confinement regime with high ion temperature the impurity convection is inwards-directed [25], as for example observed in the predecessor device Wendelstein 7-AS [26]. Up to now, however, in Wendelstein 7-X no indications for impurity accumulations have been observed, in all relevant plasma scenarios, even at high particle densities in the $1-10^{20}$ m$^{-3}$ range and in the ion-root confinement regime. It is noteworthy that recent theoretical findings predict impurity screening in a mixed collisionality regime (for hydrogen and impurity ions) at high ion temperatures [27]. More systematic studies under different plasma parameters are required to verify these highly relevant theoretical predictions. More details are found in Ref. [28] on this conference and in Ref. [24]. The eventual plasma termination by injection of impurities and the related time scales are discussed in Ref. [29] at this conference.

As outlined in Sec. 2, one of the important targets chosen for optimization is the reduction of neo-classical heat transport. This optimization is targeted for plasma with high particle density $\sim 10^{20}$ m$^{-3}$ where $T_e \approx T_i$ and the ambipolar radial electric field $E_r$ is expected to be negative throughout the plasma core, the so-called ion root. Measurements with the X-ray imaging crystal spectrometer (XICS) show a sudden transition from negative to positive $E_r$ during a step increase of the ECR heating power from 3 to 5 MW combined with hydrogen pellet injection (Fig. 7). In the time interval from $t=1.5 - 2.5$ s the line-integrated plasma density increases to $1-10^{20}$ m$^{-2}$ and the diamagnetic energy reaches a peak value of 1.1 MJ. The study of the neo-classical heat transport in the ion-root regime is in progress and results are reported in Ref. [30] at this conference.

Because of the neo-classical optimization of Wendelstein 7-X, turbulent transport is expected to play a significant role in the regulation of radial heat diffusivity and particle exhaust. Turbulence studies are therefore of key relevance and a suite of diagnostic instruments is available, namely correlation reflectometry [10], various probes for edge turbulence measurements and phase contrast imaging (PCI) for characterization of turbulence in the plasma core. PCI samples plasma density fluctuations along the line-of-sight of an infrared laser beam in the heating schemes and ECCD are found in Ref. [14] at this conference. The physics and theory of ECRH mode conversion is discussed in Ref. [21] at this conference.
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predominantly unfavourable magnetic curvature region. Nonlinear gyrokinetic simulations indicate that ion temperature gradient (ITG) and trapped electron modes (TEM) are the dominant instability mechanisms in the core plasma. Both modes are becoming unstable in the regions of unfavourable magnetic curvature, which in Wendelstein 7-X are mainly localized in the outboard bean-shaped cross section. In usual gas-fuelled discharges, ion temperature and plasma density gradient regions are well separated. Thus, ITG modes are destabilized deep in the plasma core, whereas TEM modes are localized in the edge plasma. An interesting observation is made with PCI when the plasma is centrally fuelled by pellets (Fig. 8). The density fluctuation amplitude is usually proportional to the line-integrated plasma density. However, after pellet fuelling \( t > t_d \), the fluctuation level suddenly drops. In this phase, improved confinement is observed. The fluctuation spectrogram Fig. 8 supports the picture: The entire fluctuation spectrum is strongly reduced in amplitude, but evolves transiently starting from high frequencies until the usual linear scaling is recovered. Linear gyrokinetic simulations suggest, that the turbulence suppression is the result of a radial overlap between plasma density and temperature gradients, as it is generally observed in pellet discharges. The density gradient stabilizes ITG, whereas TEM is stabilized by the temperature gradient. This finding shows that indeed turbulent transport plays an important role in Wendelstein 7-X and that its reduction, e.g., by centrally peaked density profiles, is key for the development of improved confinement scenarios. A more comprehensive discussion of turbulence and the related transport is found in Refs. [20, 28] on this conference, guided and accompanied with theory and simulation studies reported in Refs. [31, 32] on this conference.

5. ISLAND DIVERTOR OPERATION

Evidently, the installation of the 10 island divertor modules has dramatically improved the discharge behaviour and the plasma performance of Wendelstein 7-X. In the standard magnetic configuration, stable (several seconds) and full power detachment across all 10 discrete island divertor modules has been observed in a hydrogen pellet fuelled plasma at 3 MW heating power and line-averaged density of \( 2 \times 10^{19} \) m\(^{-3} \). Fig. 9 shows the time trace of the plasma and discharge parameters that illustrate the detachment event. Pellet injection leads to a step-wise density increase to about \( 5 \times 10^{19} \) m\(^{-3} \), simultaneously the central electron temperature drops to 1.5 keV. At the transition into the detached phase (at around \( t=2.5 \) s), the local peak heat flux drops from about 5 MW/m\(^2\) to less than 0.4 MW/m\(^2\). The detachment event is observed to have only little effect on the diamagnetic energy \( W_{dia} \) and the confinement time, which are both slightly deteriorated by about 10%. During the power detached phase, a highly radiative mantle was observed in the vicinity of the separatrix (the respective tomographic reconstruction is not shown here). The total radiation fraction measured with

Figure 8 Temporal evolution of the line-integrated plasma density (blue) and the turbulent density fluctuation level (green) for a pellet-fuelled discharge and the related spectrogram of plasma density fluctuations. From Ref. [28].

Figure 9 Pellet injection at \( t=1.5 \) s leads to divertor power detachment at \( t=2.5 \) s. From Ref. [18].
the bolometers is close to 100% during detachment. Infrared cameras observe the surface temperature of the divertor modules’ target elements and baffles. Fig. 10 shows the heat flux on a lower divertor module during the attached ($t=0.169$ s) and the detached phase ($t=2.73$ s) of the discharge shown in Fig. 9. The strike line pattern on the target module is clearly visible in the attached phase Fig. 9(a). The heat flux peaks at about 5 MW/m$^2$ but the major part of the wetted area has a heat flux in the range 1.5-2.5 MW/m$^2$. In the detached phase Fig. 10(b), the heat flux essentially drops to the lower resolution limit of the infrared cameras. The time evolution of the heat flux along a selected strip of the upper and the lower divertor target is shown in Fig. 10(c), demonstrating the up-down symmetric and simultaneous reduction of the heat flux by power detachment. An overview of the divertor physics is given in Ref. [18] at this conference. Further details on the heat and particle flux on the divertor are found in Ref. [33] at this conference. Power exhaust control by impurity seeding is discussed in Ref. [34] at this conference. An approach to deal with transient heat overloads on divertor targets is presented in Ref. [35] at this conference. The scrape-off layer is characterized in Ref. [36], complemented with numerical simulations in Ref. [37] at this conference.

Feed-forward control of the divertor strike lines is possible by ECCD used to tune the $\psi/2\pi$ in the plasma edge region. Fig. 11(a) shows a case study for the compensation of the toroidal equilibrium (mainly bootstrap) current of about 4 kA by counter-ECCD. Note that the sign of the toroidal current in the reference discharge without ECCD (black) is negative since the experiment was performed with reversed field. In the plasma discharge with counter-ECCD (red), all plasma parameters are the same, except that the two gyrotrons used over the full length of the discharge were operated at a launch incidence angle of 1.5°. The net toroidal equilibrium current is compensated to values close to zero. In this case, the divertor target strike lines stay fixed for the entire discharge, as indicated in Fig. 10(b). For comparison, the strike line pattern of the discharge without ECCD with 4 kA toroidal current at $t=25$ s is shown in Fig. 11(c). The red dashed line in Figs. 11(b) and (c) is the best fit to the strike lines with zero net toroidal current. The expected shift of the strike line by ~10-20 mm is clearly observed. This example shows that ECCD can be developed as a powerful tool for strike line control, especially at high electron temperature and low density, where the best current drive efficiency is achieved. The divertor target strike lines can also be using dedicated control coils underneath the divertor and external trim coils. This is discussed in Refs. [18] and [33] at this conference with special emphasis on symmetrisation of the divertor loads.

6. CONCLUSIONS

In conclusion the plasma performance of the optimized stellarator Wendelstein 7-X has significantly improved after the installation of the graphite island divertor and the graphite wall elements. Impurity and heat exhaust are now well under control and long high-density discharges became accessible, especially after boronization. With a boronized wall, the radiative density limit could be shifted to three times higher values, on the expense of a slight degradation of the energy confinement time at the limit. With relatively modest ECRH heating power, record values for the triple product in stellarators where achieved, albeit only transiently for a short time. The main limitation for high-performance plasmas is currently the limited heating power available. Nevertheless, good
plasma performance was achieved with $\beta \approx 1.2\%$ and $\beta_0 \approx 3.5\%$, mainly by using overdense ECR plasma heating in the O2 mode. The island divertor concept seems to work very well; power detachment was observed and further divertor detachment physics elements start to become evident. The divertor strike line position can be controlled either by dedicated control coils or with ECCD. The results presented in this paper are very recent and can only give a first impression of the well-controlled discharge behaviour of Wendelstein 7-X. They are however encouraging and directly relevant for the development of high-performance steady-state plasma scenarios, which is the main mission of Wendelstein 7-X.

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