W7-AS Contributions to the 11th Topical Conference
"High-Temperature Plasma Diagnostics"
(12-16 May 1996, Monterey, California)

W7-AS Contributions to the 12th International
Conference on Plasma Surface Interactions
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Controlled Fusion and Plasma Physics
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IPPIII/212 August 1996

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Die nachstehende Arbeit wurde im Rahmen des Vertrages zwischen dem
Max-Planck-Institut für Plasmaphysik und der Europäischen Atomgemeinschaft über
die Zusammenarbeit auf dem Gebiete der Plasmaphysik durchgeführt.
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W7-AS Contributions to the 11th Topical Conference
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Charge Exchange Recombination Spectroscopy
on the Stellarator W7-AS

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ABSTRACT
In the advanced stellarator W7-AS, active charge exchange recombination spectroscopy CXRS is performed on intrinsic plasma impurities. Measurements are done on the doublet lines of the singly charged species of He II, B V, C VI and N VII. A pulsed diagnostic neutral beam injector excites CX spectral lines in the visible or near UV region. By switching the beam off and on, the electron impact excited spectral light intensity from the cold plasma edge can be subtracted from the signal of interest. The spectral lines are observed after transmission by imaging mirrors and glass fibers by means of Czerny-Turner monochromators with high spectral resolution of down to 0.3 Å per spectral channel. A turnable mirror device allows for the observation at well defined spatial locations within the plasma. As detectors, photomultiplier arrays and CCD camera systems are used. By spectral measurement of the CX Doppler line intensity, line shift and line broadening we evaluate the impurity density and rotation velocity profile and the impurity ion temperature. These parameters provide the transport relevant radial electric field and the ion temperature profile. The radial ion particle and heat transport together with the ambipolar electric field play a decisive role for the optimization of the 3-D stellarator plasma.
INTRODUCTION
The advanced stellarator Wendelstein W7-AS 1 is a net current free toroidal plasma experiment. Major aim is the investigation of plasma properties in a stellarator which shows reduced Pfirsch-Schlüter current and Shafranov shift in comparison to a classical stellarator. With respect to a future fusion reactor, the stellarator concept is characterized by steady state operation capability without the menaces arising from disruptions, as they can occur in tokamaks. Fusion relevant plasma parameters are the ion temperature $T_i$, the energy confinement time $\tau_E$ and the plasma ion density $n_i$. For the assessment of present plasma experiments the measurement of these key parameters is of primary interest.

In W7-AS, active charge exchange recombination spectroscopy CXRS 2 is used to gain access to some of these parameters. $T_i$ can be measured directly by CXRS. The confinement properties of a toroidal plasma, especially the parameter $\tau_E$, are in general not accessible to a direct measurement. However, CXRS allows for the experimental evaluation of the transport relevant radial electric field $E_r$ which can have strong impact on $\tau_E$. Field and radial particle transport are closely linked to each other. The radial particle fluxes determine the size and radial profile of the field, and the field determines in turn the radial fluxes. Under long mean free path conditions in stellarators the 1/ν transport regime is expected to occur, with strongly enhanced radial transport for the case without field. That unfavourable effect, however, is considerably compensated according to neoclassical theory for the case that a radial electric field is taken into account, for the particle and the heat transport coefficients. Therefore the field plays a crucial role for stellarator confinement. The field impact on the transport coefficients is much stronger for the ions than for the electrons. As the particle confinement properties of W7-AS are in general determined by the radial ion transport, this emphasizes again the importance of the field.

Additionally, CXRS can measure impurity density profiles of intrinsic light plasma impurities like helium, boron and carbon. These experimentally determined profiles are compared to the results of neoclassical impurity transport calculations. Thus, the impurity particle confinement properties of W7-AS are determined.

MEASUREMENT PRINCIPLE
The neutral particle beam of a pulsed diagnostic injector is used to excite CX radiation in the visible or near UV range. That beam is modulated to allow for the subtraction of the electron impact excited background radiation and CX light from the colder plasma edge. The beam power is kept small enough to avoid plasma
heating. The CX light level is very low and we have to integrate the light intensity
for at least 50 msec during the active beam phase for a sufficiently high signal to
noise ratio. Further 50 msec are needed for the phase with beam off. Thus, the
temporal CXRS resolution is restricted to about 10 Hz. The beam accelerating
voltage can be varied between 18 and 36 kV. Thus an optimum beam energy can be
chosen such that we have a maximum CX emission cross section for the impurity
species under investigation on the one hand, and maximum beam penetration into
the plasma on the other. Spectroscopic active Hα measurements of the three
energetic beam components in cold background gas reveal that about 50% of the
beam neutrals have the full energy, and about 25 % are in the one half and 25% in
the one third energy component, respectively. In parallel to CXRS, CX neutral
particle energy analysis NPA is performed in the diagnostic beam, a technique
which also can provide Tj profiles.

The dominant intrinsic impurity in W7 AS is boron, which is present as result of
the boronization of the vessel wall. Carbon is found because graphite rail and inner
limiter are installed in W7-AS. Helium can be used for CXRS because cleaning
He glow discharges are performed from time to time to improve the wall recycling
conditions. However, because that background He content is small and varies from
shot to shot, sometimes additional external He gas puff is applied at the beginning
of the discharges such that a concentration of about 1% - 5% is achieved. This
technique provides a constant and reproducible He content even for long series of
discharges. He for CXRS combines several advantages: it is cheap, easy to handle,
has strong lines in spectral regions without disturbing background lines and its
correction to the effective charge state Zeff is small. Nitrogen is applied as
external gas puff for edge cooling experiments into the chain of magnetic islands
near the plasma edge for iota(a) = 1/2, and can also be used for CXRS.

For all investigations we focus on the following doublet lines of fully ionized
impurity ions which are known from our measurements to be free from disturbing
spectral background lines:

\[ \text{He}^{2+} + \text{H} \rightarrow \text{He}^{+} + \text{p} + \text{hv} \]
[4687 Å: \(4f^2{^2F_0} (5/2,7/2) \rightarrow 3d \, \text{^2D} (3/2,5/2)\) ]

\[ \text{B}^{5+} + \text{H} \rightarrow \text{B}^{4+} + \text{p} + \text{hv} \]
[2982 Å: \(6h \, \text{^2H}^0 (9/2,11/2) \rightarrow 5g \, \text{^2G} (7/2,9/2)\) ]

\[ \text{C}^{6+} + \text{H} \rightarrow \text{C}^{5+} + \text{p} + \text{hv} \]
[3434 Å: \(7i \, \text{^2I} (11/2,13/2) \rightarrow 6h \, \text{^2H}^0 (9/2,11/2)\) ]
\[ \text{N}^7_+ + \; \text{H} \; \Rightarrow \; \text{N}^6_+ + \; p + \; \text{hv} \]

[3887 Å: 8K^p (13/2,15/2) \Rightarrow 7I^2 (11/2,13/2)]

Here, \( \text{H} \) denotes fast neutral beam particles, \( p \) the protons after the CX reaction, \( \text{hv} \) the CX spectral light. For each reaction, the corresponding CX spectral transition is given in brackets together with the wavelength.

To evaluate the plasma parameters of interest, the spectral CX line shape is measured as a function of time and the radial position in the plasma. Thus, the CX spectral line intensity, the Doppler line broadening and the line shift become accessible after background subtraction.

From the CXRS line intensity one can determine the density profile of one completely ionized impurity species if the neutral density in the diagnostic beam is known. For that sake, the beam attenuation in the plasma is calculated from a recursive algorithm \(^7\). Recursion is necessary because the attenuation depends also on the unknown profile itself. CX and ionization with the background plasma, all intrinsic impurities and the impurity species under investigation is taken into account for the calculation of the attenuation. For the first iterative loop, an impurity model profile shape is assumed, which converges for subsequent iterations into a final profile. As our measurement system is not calibrated in absolute light intensities, only a density in arbitrary units is obtained. Estimates of absolute impurity densities are deduced for calibrated external gas puff of light impurities. For that case combined CXRS measurements, \( Z_{\text{eff}} \) from Bremsstrahlung and absolutely calibrated vacuum ultraviolet spectroscopy VUV \(^8\) are performed.

The radial \( T_i \) profile is determined from the CX Doppler line broadening by the following procedure. First the expected Zeeman spectral line splitting is calculated according to the local magnetic field in the plasma. The polarization properties of the monochromator together with the grating are measured with a calibration lamp and are taken into account for the relative attenuation of each Zeeman component separately in the monochromator. Thus, an individual Zeeman pattern for each spectral transition, radial plasma position, magnetic field geometry and impurity species is developed. Additionally, spectral line broadening by finite monochromator slit widths is calculated. Spectral line convolution by the monochromator instrument function is directly measured with cold, low pressure Hg spectral line lamp and is also considered. That procedure allows for the development of a numerical fit model for the spectral line shape. As free fit parameters a constant spectral background, the total line intensity, broadening and shift are left. A modified Levenberg-Marquardt algorithm from the commercial
IMSL software library is then used to minimize the deviation between the measured spectral line shape and the calculated model. That procedure provides also the value of the impurity rotation velocity from the measured spectral CX Doppler line shift. The rotation can be enhanced into toroidal direction by momentum input from the neutral beam injection heating NBI, and is damped by counter-acting viscosity. The transport relevant radial electric field $E_r$ is closely coupled to toroidal and poloidal rotation velocities. The ratio of the respective velocity components depends on the magnetic field geometry. Both components are registered in W7-AS by means of two different CXRS systems, which are described below.

**CXRS SYSTEMS**

The CX light intensity is registered by two systems. The first one is situated such that the observation chord is within one poloidal plane of W7-AS, almost perpendicular to the magnetic field lines. A turnable mirror allows to pivot the chord within the poloidal plane. Thus different radial positions in the beam are chosen. The spatial resolution in radial direction is limited to about ±5 mm. The CX light is guided via an 18 m long imaging mirror system with 9 single mirrors directly onto the entrance slit of a 1.25 m Czerny-Turner grating monochromator with a 2400 mm$^{-1}$ grating. In the exit slit plane of the monochromator, a linear array of 15 quartz fiber bundles is installed, such that 15 spectral channels can be measured simultaneously. The single fibers of each bundle in the exit plane are linearly arranged with a slight curvature to compensate for the curvature of the image of the straight entrance slit. Each quartz fiber bundle is connected to one photomultiplier tube. The total slit height of 50 mm is used to collect the maximum possible light intensity. In that arrangement, we have a spectral resolution of 0.3 Å per channel in first order observation. A poloidal cross section of the plasma, together with the diagnostic beam and some observation chords is shown in the fig. 1. The plasma measures, from the bottom to the top, about 64 cm. As the observation chords are here almost perpendicular to the magnetic field lines, that system allows for the observation of the poloidal impurity rotation.

The second system is positioned such that we have an angle of about 50 degrees between the magnetic field lines and the observation chords. An f=40 mm achromatic lens system images the CX light onto a rectangular quartz fiber bundle. The spatial position of the bundle can be changed, thus changing the inclination angle of the observation chord and the observed radial position in the diagnostic beam. The minimum spatial resolution in radial direction is here limited to about ±
2.2 cm. The fiber bundle is 6 m long and is plugged directly onto the entrance slit of a 0.5 m Czerny-Turner monochromator with a 2400 mm\(^{-1}\) grating. In the plane of the entrance slit the single fibers in the bundle are arranged linearly with a slight curvature, again to compensate for the curvature of the image of the entrance slit in the plane of the exit slit. In front of the bundle is a mechanical slit which can be used to enhance the spectral resolution. The full monochromator slit height of 20 mm is used to collect as much light as possible. The spectral resolution is about 0.3 Å / pixel for the case of a 30 μm wide mechanical entrance slit in first order observation. The detector is a 512 X 512 pixel CCD camera system with Peltier cooling. An image intensifier is used as fast shutter to allow for an exact synchronization of the camera readout in phase to the diagnostic beam duty cycles. Due to the observation geometry, the second system allows only for the conjunctive measurement of the projections of the toroidal and poloidal rotation component onto the observation chord. As the geometry of both systems is known as well as the purely poloidal velocity component from the first system, the purely toroidal velocity component can be calculated.

The CX light fluxes are rather low; typically we measure active photon fluxes of about 10\(^5\) photons per second for the CX spectral line under investigation. Therefore our signal to noise ratio S/N depends strongly on the neutral beam density and on the impurity density in the plasma. Typically, S/N is between 1 and 5.

**RESULTS**

Impurity density profiles are measured for light impurities, which are then compared to results of neoclassical impurity transport codes. Figure 2 shows a He\(^{++}\) density profile measured with the first spectroscopic system, together with the result of the neoclassical SITAR (Simulation of Impurity Transport and Radiation) code calculation. The absolute He density is fixed for the calculation to 2% of the electron density, which is consistent to absolutely calibrated VUV results. Spectroscopic measurement and neoclassical calculation are in good agreement, indicating the dominating role of neoclassical particle transport for impurities in W7-AS. The measured profile is slightly more hollow than the result of the calculation. This effect is assumed to be provoked by a small positive \(E_r\) near the plasma center, which leads to an enhanced outward convection for the impurities because of their high charge state. Including that field value in the neoclassical calculation reproduces the measured profiles much better and confirms
thus our assumption. In general we find for W7-AS that the profile shape of $n_e$ and the He density are quite similar.

The error bars result from the statistical measurement uncertainty of each individual spectral point, and from the uncertainties of the calculation of the neutral beam attenuation by the background plasma. To determine that error, the measurement errors of the Thomson scattering $n_e$ profile are taken into account in the recursive algorithm mentioned above, because $n_e$ is the decisive plasma parameter for the beam attenuation.

Figure 3 shows a result of a $T_i$ profile measured with the second spectroscopic system on He++. The high central $T_i$ of about 1.25 keV is achieved by combined heating with NBI and ECRH. Thus, a high heating power (nominal total heating power = 1.8 MW) together with a relatively small electron density ($n_e (0) = 6 \times 10^{19} \text{ m}^{-3}$) is achieved. The measured points for $r > 15$ cm stem from passive measurements of the Doppler line broadening of the electron impact excited triplet line of B IV at a wavelength of 2824 Å. In that radial region, in general the active CX signal is too small to provide a reasonable signal to noise ratio and therefore passive measurements are performed here.

Figure 4 shows a result of an $E_r$ profile, also measured on He++. The field is determined from the radial force balance equation, which employs the measured ion pressure gradient term for the impurity under investigation, and the rotation velocity profile. The ion pressure, in turn, is provided by the impurity density profile, as shown as an example in fig. 2, and the corresponding ion temperature profile, as shown as an example in fig. 3. We find that the major contribution to $E_r$ comes from the poloidal rotation. The ion pressure gradient contributes typically to less than 10%, the toroidal rotation to less than 2% because of the large ratio $B_\theta/B_\varphi = 20 - 100$, and because the toroidal rotation velocity is much smaller than for a comparable tokamak, as a consequence of the lack of toroidal field symmetry in a stellarator. To the pressure gradient term, \( \nabla \langle T_i \rangle \) contributes to 90%, \( \nabla \langle n_i \rangle \) to 10%. The measured $E_r$ are compared to neoclassical DKES transport code calculations. For that sake, the radial particle fluxes are calculated by DKES as function of $E_r$. Then the ambipolarity constraint is solved to search for stable „root solutions“ of the field. Typically, we find experimentally an $E_r$ profile which is consistent to the „ion root“ solution from the neoclassical calculation, which is in general accompanied by negative $E_r$. Some Langmuir probe measurement points outside the plasma are also shown in fig. 4. The field maximum at $r \approx 12$ cm is located at the radial position of the gradients, indicating the correlation between
large field and reduced radial transport. Large fields and large density and
temperature gradients co-incide and support each other.

Figure 5 shows an example for the measured toroidal rotation profile for a
discharge with 3 NBI injectors for unbalanced co-injection with a nominal heating
power of 1.2 MW. As expected from calculated power deposition profiles \(^{16}\), the
toroidal velocity is maximum near the plasma center and decreases for increasing
minor radius. The maximum value of about 65 km/sec is higher than earlier results
presented elsewhere\(^{13}\), presumably because now the toroidal magnetic field ripple
is smaller, and thus the rotation damping effective toroidal viscosity also.

CONCLUSIONS
Radial profiles of the impurity density for light impurities, the ion temperature, the
poloidal and toroidal rotation velocity, and the radial electric field are measured by
CXRS in W7-AS. Spatial and temporal resolution is achieved by use of a
diagnostic neutral beam injector. The measurements are compared to the results of
numerical neoclassical transport codes, yielding consistency between the
experimental observation and the calculations. Thus the validity of neoclassical
particle transport theory for W7-AS is drawn out.

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FIGURE CAPTIONS

Fig. 1 - Sketch of one poloidal cross section of the W7-AS plasma at the toroidal position of CXRS observation. The tilted ellipses show the magnetic surfaces. The thick broken ellipse indicates the last closed flux surface at the radial position of the limiter. Flux surfaces outside are behind the limiter. The hatched area is the cross section of the diagnostic neutral beam. The straight lines show 5 possible observation chords.

Fig. 2 - Spectroscopically measured He$$^++$$ density profile (dots) together with a fit of a generalized gaussian function (broken line) to the measurement. The solid line shows the result of the neoclassical SITAR calculation. The absolute He density is fixed to 2% of the electron density, a value which is confirmed by VUV.

Fig. 3 - Spectroscopically measured ion temperature profile (dots) together with a fit of a generalized gaussian function (broken line) to the measurement.

Fig. 4 - Spectroscopically measured radial electric field (dots), and Langmuir probe measurement (squares). The solid line shows the result of the neoclassical DKES calculation.

Fig. 5 - Spectroscopically measured toroidal rotation velocity profile for an unbalanced NBI discharge.
Tomography by the maximum entropy method for the W7-AS and W7-X multichannel bolometer systems

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June 18, 1996

Abstract

A maximum entropy approach has been applied to the tomographic inversion of the line integrated measurements of radiation power flux by a multichannel bolometer system. This method was applied to a H-mode discharge in the W7-AS stellarator, with increasing radiation from the plasma core after the transition being observed. Also a neutral beam injection discharge was investigated, and the time evolution of a marfe was documented. This method has also been applied to the preliminary design of a multi-camera system for full tomographic reconstruction in the W7-X stellarator. In addition, measurements with wire mesh shielded metal foil bolometers have been carried out and the shielding of these bolometers from microwaves produced by the 140 GHz gyrotrons used for electron cyclotron heating (ECRH) has been demonstrated.
1 Introduction

The maximum entropy method has previously been used in plasma physics for Abel inversion and spectroscopic deconvolution [1]. The entropy in this case is a measure of the information content of a sample [2]. A self-contained review of the ideas of Bayesian probability theory, its realization by application of variational calculus and a derivation of the standard formalism behind the principle of maximum entropy can be found in the literature [3].

A description of the numerical methods of maximum entropy is presented in section 2. The experimental results from the W7-AS stellarator [4] are presented in section 3 and the time evolution of the radiation power in a H-mode, and an NBI discharge in 2 dimensions is reconstructed. During an ECRH pulse it was shown that bolometer shielding by wire mesh is effective in screening direct microwave radiation from the 140 GHz gyrotrons. In section 4, the preliminary design of the W7-X multichannel bolometer array is presented and simulations considering the required accuracy of the relative calibration of the channels is discussed. The conclusions are summarized in section 5.

2 Numerical methods

In this paper, the maximum entropy method is applied to the tomographic inversion of the line integrated measurements of radiation power flux by a multichannel bolometer system. The input data and the associated error bars represent a set of linear constraints to a matrix equation of the form:

\[
I_j = \sum_{k=1}^{n} w_{jk} P_k \tag{1}
\]

where \( I_j \) is the \( j^{th} \) line integral, \( w_{jk} \) is the weighting factor of the \( k^{th} \) pixel for the \( j^{th} \) line integral and \( P_k \) is the radiation power density of the \( k^{th} \) pixel.

By taking a discrete set of \( n \) points in the radial direction of the radial profile of the radiation power density, \( P_i \), the entropy, \( S \), is given by:

\[
S = -\sum_{i=1}^{n} \rho_i \ln \rho \tag{2}
\]

where \( \rho = P_i / \sum_{j=1}^{n} P_i \) and \( S \) is maximized subject to the linear constraints [5].
For Abel inversion, the radiation power profile, \( P_i(r) \), is parameterized so that for all values of the fitting coefficients \( P_i \) is positive definite. In the simplest case for two fitting coefficients, this is of the form:

\[
P_i(r) = \exp(a_0 + a_1 * r^2)
\]  

(3)

with the final choice of the number of fitting coefficients finally determined by the goodness of the fit. The coefficients describing the \( P_i \) profile, and therefore the flux surface value of the radiation power density, are adjusted iteratively, comparing the measured line integrals with that calculated for each simulation so that their difference is minimized. The quantity minimized in this case is:

\[
\chi_{error} = \sum_{i=1}^{N} \left( \frac{d_i}{\sigma_i} \right)^2
\]  

(4)

where \( d_i \) is the difference between the measured and calculated value of the line integrals, \( \sigma_i \) is the standard deviation of the measurement and \( \chi_{error}/N \) is a measure of the quality of fit. Ideally the latter attains a value of 1 so that the fit lies within one standard deviation of all measurements. Concurrently, \( S \) is maximized with the result that the flattest \( P_i \) profile that achieves such a fit from the possible family of \( P_i \) profiles is chosen [1]. In the case that the maximum entropy principle was not imposed, it could arise that by solely minimizing the least squares fit without paying respect to the error bars of the measurement one could arrive at a profile with radial structure that would be unjustified.

In the case of 2-dimensional tomography, where two views of the plasma are available to allow the determination of \( m=1 \) component of the radiated power with respect to the flux surfaces (as in the case of the measurements in W7-AS), the entropy was constructed as the sum over two \( P_i(r) \) profiles and a free parameter allowing the maximum along a radius to lie at an arbitrary angle with respect to the minor axis.

For the simulations on W7-X, where 10 views were assumed to be available, the sum shown in equation 2 was used with the radiation power density values at each pixel on a 32x32 grid rather than each flux surface value.
3 W7-AS experiments

On the W7-AS stellarator two different types of bolometer systems are in operation. There is a two camera system with 30 channels per camera using 5 μm stainless steel foil mounted on an 1.5 μm MgO insulator with oxygen doped germanium as the thermistor. A 5 μm gold grid forms the electrodes to measure the resistance of the germanium layer which changes with the temperature of the foil which in turn is a function of the incident power flux of radiation from the plasma [6]. With reference to Fig. 4, the cameras view the triangular flux surfaces through a slit at symmetric positions above and below the midplane at R = 240 cm and z = ±36 cm.

A single camera of 8 channels using 4μm gold foil mounted on an 7.5 μm Kapton insulator and a bridge of 4 gold meander resistors on a measurement and reference arm are also currently being tested on W7-AS [7, 8]. These bolometers are in routine operation on JET, ASDEX-Upgrade and Alcator C-mod. A wire mesh with dimensions 13 μm x 94 μm has been installed in front of the absorbing metal foil of these bolometers. Microwave power produced by the 140 GHz gyrotrons used for electron cyclotron heating (ECRH) and unabsorbed by the plasma induces unwanted heating of the absorbing foil as shown in Fig. 1. The effectiveness of the shielding is also clearly demonstrated. As W7-X will depend strongly on ECRH heating, such a demonstration is necessary for confidence in the successful operation of a multichannel bolometer system on this machine.

The H-mode on W7-AS has been observed previously in ECRH discharges [9]. In the discharge considered here, a H-mode transition could be produced with 340 kW of absorbed NBI power at a the rotational transform, ρ, of 0.525 as shown in Fig. 2. It has been documented that the H-mode in the W7-AS stellarator is found only in a narrow ρ range between 0.51 to 0.53, and this is attributed to the long connection lengths which exist in this ρ range [10]. In Fig. 3, the line integrated measurements and the resulting fit by the maximum entropy method after summing the lines of sight over the radiation power profile are shown. In Fig. 4 the reconstructed radiation power density before and after the H-mode transition are shown. Clearly the radiation from the core increases after the H-mode transition, consistent with this previously observed feature in tokamaks [11, 12].

A marfe has been observed in a discharge with 1.8 MW NBI power and rising density in the absence of gas puffing. For this discharge, with an ρ of 0.564, no H-mode transition is expected. The marfe, characterized by a
localized region of intense radiation, moves away from the outwardmost tip of the triangular flux surface at a radius in the vicinity of the last closed flux surface towards the bottom of the machine and back again over a time scale of 150 ms. Shown in Fig. 5 are the line integrated measurements and the resulting fit by the maximum entropy method is shown. In this case, the fits lie well outside the measurement error bars. However, the error bars used for fitting were 10% of the measured value, reflecting the difficulty of obtaining a satisfactory relative calibration factor for these bolometers. Further improvements to the fit will be sought by allowing for a variable thickness of the outermost flux surface. The general features of the measured line integrals are reproduced by the fit and clearly indicate the presence of an intense localized radiation source that moves at the plasma boundary. Shown in Fig. 6 is the 2-dimensional tomographic reconstruction at the beginning, where the marfe is positioned at the outwardmost tip of the triangular flux surface, and at the maximum intensity of the event, where the marfe is positioned at the bottom of the machine between the tips of the triangular flux surface.

4 W7-X simulations

The maximum entropy method has also been applied to the preliminary design of a multichannel bolometer system for the W7-X stellarator [13]. A 10 camera system with 30 channels per camera has been simulated with a view to full tomographic reconstruction of the radiated power. The number of cameras was chosen in accordance with the m=5 island structures present in the flux surfaces.

A radiation power density was assumed and line integrals of the incident power flux for each bolometer was calculated. The tomographic inversion of these line integrals in this case was carried out on a 32x32 pixel grid with or without information about the last flux surface. It is envisaged that the anisotropic diffusion model can be adapted to incorporate knowledge of the flux surface orientation at each pixel [14], forcing a degree of coupling on the radiation power density of neighbouring pixels. A sensitivity study was carried out to determine the required accuracy on the relative calibration of each line of sight. A random error of 5% was added to the simulated line integrals and was sufficient to seriously deteriorate the reconstructed radiation power density.
Of particular concern, is the validity of the measurements under conditions of high neutral gas density (e.g. in the vicinity of a divertor), where a component of the incident power on the bolometer is due to localized neutral particles from charge exchange at the plasma edge [15]. In this case the power is radiated asymmetrically so that a degree of error enters into the line integrated measurements. A possible solution could be attempted by making a first order correction to the bolometer measurements. By coupling the bolometer measurements to a neutral gas/charge exchange simulation code, the measurements for the calculated component of incident power due to charge exchange losses could be numerically corrected.

5 Conclusions

The maximum entropy method has been applied to the two dimensional tomographic inversion of line integrated bolometer measurements from the two camera system in the W7-AS stellarator. An increase of core radiation with time after a H-mode transition, consistent with this feature in tokamaks, has been observed. A marke in a NBI discharge, with the movement of an intense localized source of radiation over a time scale of 150 ms, has been documented. The successful shielding of metal foil bolometers in an ECRH environment has demonstrated that this diagnostic will be able to deliver reliable measurements on W7-X.

The preliminary design of a multichannel bolometer system for full tomographic reconstruction of the radiated power density in W7-X has been presented and the restriction on the relative calibration of the channels to realise this aim has been discussed. In particular, the component of incident power on the bolometer from energetic neutral particles generated by charge exchange in the plasma edge will need to be modelled to provide line integrated measurements with sufficient relative accuracy for reliable tomographic reconstruction.
References


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DIAGNOSTICS FOR W7-X

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Abstract: The Wendelstein 7-X (W7-X) experiment is a concept test for properties of reactor relevant plasmas in advanced stellarators. Prominent features are a modular superconducting coil assembly, a 5-fold toroidal symmetry and a helical magnetic axis. Due to the optimisation process W7-X is characterized by a vacuum magnetic field configuration with smooth magnetic surfaces, improved equilibrium properties with small dependence of rotational transform and shear on the plasma pressure $\beta$, good MHD stability properties due to magnetic well stabilization, reduced neoclassical transport losses and negligible bootstrap current in the LMFP regime, good collisionless $\alpha$-particle confinement in an equivalent reactor, and -as a technical aspect- good feasibility of the superconducting modular coils. W7-X will be heated by ECRH (CW) and pulsed (10s) NBI and ICRH. The envisaged parameters are $T_e \leq 10$ keV, $T_i \leq 6$ keV central densities $\leq 3 \times 10^{20}$ m$^{-3}$ with an averaged $\langle \beta \rangle \leq 5\%$.

Despite the complicated geometrical structure, all basic diagnostics are compatible with W7-X. Generally, diagnostic methods and applications in a stellarator are not different to those in tokamaks. However, special efforts are being made to equip the experiment with those diagnostics necessary to measure the quantities directly related with the optimisation of the machine: the verification of the predicted magnetic topology and characterization of the configuration in the whole
parameter range, the identification of equilibrium and stability, and the determination of the confinement properties.

THE W7-X STELLARATOR

The Wendelstein 7-X experiment is an integrated concept test for properties of reactor relevant plasmas in optimized advanced stellarators (1-4). In order to optimize a stellarator fusion experiment ab initio an interfaced set of 3-dimensional codes was developed and has led to an optimized stellarator configuration which can be properly considered as computational stellarator. Stellarators need optimization because classical physics issues seriously limit their viability as fusion power plants. An example of their optimization potential are quasi-helically symmetric toroidal magnetic fields, which show that the magnetic geometry of a stellarator can be decoupled from its real space geometry. Thus, stellarator optimization does not primarily consist in quantitatively improving some given basic concept but in selecting basic physics properties. The W7-X configuration leads to a toroidal plasma equilibrium of the so-called HELIAS type (helical advanced stellarator). Prominent geometrical features are a modular superconducting coil assembly (50 non-planar modular field coils, 20 planar auxiliary field coils) a five-fold toroidal symmetry (5 modules of 72°), with a helical magnetic axis and, resulting in a truly 3-dimensional plasma topology with bean-shaped plasma cross sections at the positions of strongest curvature (ϕ = 0°) changing via drop-like cross sections (ϕ = 18°) into triangular ones in the regions of smallest curvature (ϕ = 36°). The rotational transform on axis τ(0) and at the boundary τ(a) can be varied from 0.75 ≤ τ(0) ≤ 1.01 and 0.83 ≤ τ(a) ≤ 1.25, respectively, with the aid of the planar coil set.

Essential characteristics due to the optimisation process are:
a vacuum magnetic field with smooth magnetic surfaces and a regular boundary which can be envisioned as a resonant island diverter

- improved finite-\(\beta\) equilibrium properties resulting from a small Shafranov shift (strongly reduced Pfirsch-Schlüter current) and a small change of rotational transform and shear with \(\beta\) due to a vanishing bootstrap current which yields a high equilibrium \(\beta\)-limit -
- good MHD stability properties due to magnetic well stabilization and suitable flux surface shaping
- reduced neo-classical transport losses in the reactor relevant LMFP regime so that the so-called \(1/\nu\)-dependence of the diffusion coefficients does not appear
- good collisionless \(\alpha\)-particle confinement (in equiv. reactor) towards higher \(\beta\)
- good feasibility of the superconducting coil set.

The major and minor radii are 5.5 m and 0.53 m, respectively, the magnetic field strength on axis \(\leq 3\) T, typically \(2.5\) T and \(1.25\) T to fulfill the resonance condition set by ECRH which will be the main W7-X heating system. In two steps the machine will be equipped with steady state (cw) 140 GHz gyrotrons with a total power of 6 MW, in the final stage 10 MW. 70 GHz tubes will be installed for 1.25 T operation and O-mode heating during start-up. The beam launching scenario and the gyrotron operation will allow for physics programs as well i.e. ECCD and power modulation experiments. Besides true steady state operation under ECRH conditions, \(H^0\), \(D^0\) injection at 55 keV with maximum pulse lengths of 10 s and at power levels of 5 MW, in final stage 20 MW, will complement ECRH. The NBI system will also allow for beam intensity modulation for fast particle studies. ICRH will complete the heating scenario with 3 MW at the beginning and 8 MW total power in the final stage. Flexible connections to various antennae are planned as well as modulation capability for thermal and non-thermal transport studies. The envisaged plasma parameters in W7-X under these heating conditions are: \(T_e \leq 10\) keV, \(T_i \leq 6\) keV, densities up to \(3 \times 10^{20} \text{ m}^{-3}\) and average \(\langle \beta \rangle\) values of up to 5%.
The power load onto the divertor tiles under maximum power conditions will not exceed 10 MW/m². The cw heating by ECRH will allow for studies of neo-classical transport in the LMFP regime while NBI at low magnetic field will be used for $\beta$-limit investigations.

Important aims of the W7-X experiment are studies of energy and particle exhaust in a HELIAS configuration under quasi-stationary conditions and the development of a reactor-relevant divertor system. Scenarios to be investigated will include operation with radiative boundaries. The divertor concept for W7-X exploits the inherent field line diversion property of optimised HELIAS configurations, comprising the variants of ergodic and island field line diversion, also utilising additional loops for island and divertor control.

**W7-X DIAGNOSTICS, GENERAL DEMANDS**

In W7-X standard diagnostics, as known from other stellarators, torsatrons or heliotrons and tokamaks, will be used to measure basic plasma quantities (5,6). Thomson scattering ($T_e$, $n_e$), ECE ($T_e$), reflectrometry ($n_e$, gradient), interferometry / polarimetry ($n_e$), CX-neutral analysis ($T_i$), CXR-spectroscopy ($T_i$, $n_z$, $v_{tor+pol}$, $E_r$), impurity spectroscopy ($n_z$, $\Gamma_z$), $Z_{eff}$, bolometry ($P_{rad}$), $H_{\alpha}$ arrays (recycling), soft-x and Mirnov loops ($T_e$, MHD), diamagnetic and flux loops (energy, flux to assess the reduction of the PS-currents), edge probes ($n_e$, $T_e$, $\phi$), calorimetry (heat load), fast Li-beam ($n_e$, edge), plasma and IR video (plasma configuration and monitor, target loads) plus neutron and fluctuation diagnostics form the basis foreseen for the scientific studies planned. Important for ECR-heating and ECE diagnostics is the 1/R magnetic field dependence (equivalent to tokamaks) in the bean shaped cross sections of W7-X which allows to exhaust the full potential of EC absorption and emission properties of an optically thick plasma i.e. well localized power deposition and high resolution electron temperature diagnosis.
A main goal of W7-X is the experimental verification of the major design elements, selection of \( \text{iota}=1 \) with moderate shear, improved equilibrium, drift optimization, energetic particle confinement, negligible bootstrap current, and the role of the drift optimization on anomalous transport, for which the diagnostics must be specially adapted to. The diagnostic set must as well be capable of providing the necessary information in experiments.

- demonstrating the high performance of the machine in various confinement scenarios i.e. high \( \text{n}T_\parallel \), high \( T_e \) and/or \( T_i \), high \( \beta \), \( \beta \)-limitations, high density operation, in L- and H-mode and with safe power handling
- making use of stellarator specific properties like long pulse operation, low shear conditions, especially for island studies, and radial correlation length investigations
- under operation conditions with and without EC driven currents for MHD measurements and all studies on the tokamak-stellarator comparison without loosing accurate knowledge of the configuration which is expected to be stable close to the vacuum configuration.

Because of the reduced toroidal symmetry compared to a tokamak, transformation of the local laboratory coordinates of individual diagnostic sightlines and observation volumes to flux coordinates is essential. Experience from W7-AS shows that mapping of points or lines of sight of one diagnostic in a specific poloidal plane along field lines to other diagnostics and plasma cross sections is possible with sufficient accuracy due to the well known structure of the magnetic field. Generally, this mapping provides the toroidal and poloidal correspondence of measured quantities which are expected to be constant on magnetic surfaces. It provides profiles of different diagnostics as a function of an effective (flux conserving) radius, it allows tracing of information along field lines and gives the radial location of line-integrated measurements after Abel inversion and tomographic reconstruction.
In order to provide a set of basic diagnostics at startup of the machine and to prepare for a later stage of diagnostic needs and developments possible and desirable diagnostics have been categorized in three levels:

- level-1 comprises the basic diagnostics for machine and divertor operation and plasma characterization as given above.
- level-2 extends the diagnostic potential of the level 1 set with respect to additional parameters and / or higher spatial and / or temporal resolution, redundancy, poloidal / toroidal asymmetries, correlations etc.
- level-3 finally includes very special advanced diagnostics which may be of importance for special investigations in a well diagnosed machine and which may include very recent trends and developments towards new diagnostic possibilities: extensive diagnostics for fluctuations in E-fields, density, temperature as well as correlations between these quantities.

Design Elements Flux Topology and Optimization concerning Stability and Equilibrium

Extended magnetic and soft x-ray diagnostics are needed to address the optimisation of the configuration with respect to equilibrium, stability and pressure driven currents. On the predecessor experiment W7-AS a very sophisticated soft x-ray diagnostic has been used extensively to investigate and document the stability and equilibrium of its partially optimized configuration. This plasma core diagnostic is complemented by probe arrays and video diagnostics which image the edge topology of the plasma including islands, as already established for W7-AS. In detail, the verification of the predicted magnetic topology is based on:

- tomographic reconstruction of the x-ray emissivity
- characterization of different configurations in the whole flexibility range
- investigation of high-$\beta$ operation at reduced magnetic fields
- deduction of axis position, separatrix position and Shafranov shift
- measurement of the dipole field emerging from the Pfirsch-Schlüter-currents
- verification of low bootstrap current its parameter dependencies and the flow direction
- characterization of the edge topology
- effects of pressure induced magnetic islands
- relation to vacuum field line mapping and transformation codes

Equilibrium and stability identification require:
- detection of rational surfaces (MHD activity, stationary islands)
- identification of radial and poloidal mode structures
- operational limits or enhanced transport due to MHD instabilities
- verification of predicted instability thresholds and identification of instability mechanisms
- establishment of unfavourable (more unstable) configurations utilizing the 20 auxiliary coils to investigate thresholds and to show effects of stellarator optimization

The low shear allows to address stability and turbulence issues like fluctuation and correlation measurements. These studies will profit from cw operation. Rotational transform = 1 in the plasma core will be of specific interest along with the stability issues around the critical 9/10 and 11/10 resonances. In particular, the different scientific studies will be based on measurements with the diagnostics as given in table 1.
<table>
<thead>
<tr>
<th>Magnetic Topology</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vacuum field topology</td>
</tr>
<tr>
<td>configuration, plasma axis, Shafranov shift</td>
</tr>
<tr>
<td>PS- and bootstrap currents, flow direction</td>
</tr>
<tr>
<td>edge topology</td>
</tr>
<tr>
<td>pressure induced islands, high-beta effects</td>
</tr>
<tr>
<td></td>
</tr>
<tr>
<td>Equilibrium and Stability</td>
</tr>
<tr>
<td>rational surfaces, MHD activity, stationary islands</td>
</tr>
<tr>
<td>radial/poloidal mode structure</td>
</tr>
<tr>
<td>operational limits, instability threshold</td>
</tr>
<tr>
<td>unfavourable configurations, thresholds, optimization</td>
</tr>
</tbody>
</table>
Design Elements Classical Confinement and Neo-Classical Transport

Neo-classical transport is a threat to stellarator reactors since particles trapped in local magnetic mirrors can lead to rapid losses. This is especially true in the collisionless LMFP regime where, as mentioned above, this neo-classical effect leads to the so-called 1/ν-regime. However, approximating quasi-helical symmetry this regime does not appear.

The key element in these losses is the radial drift of the trapped particles away from magnetic surfaces. This drift is minimized if the particles are localized in regions of small poloidal field variation. In a standard stellarator it is the helical ripple which leads to the enhanced losses. In the LMFP regime trapped particles dominate neo-classical transport. It is mainly the synergetic effect of the leading Fourier harmonics of a HELIAS configuration which leads to the reduction of the effective helical ripple (by a factor of 10-20 in the 1/ν-regime) and radial transport.

For the ions in the 1/ν-regime, however, the radial electric field is more effective in reducing the radial diffusion than the magnetic drift. The measurement (and control?) of the radial electric field is therefore of special importance.

Investigations on neo-classical transport properties will also benefit from the high configurational flexibility of W7-X whose mirror ratios can be varied in a wide range and where its influence on trapped particle fractions and neo-classical damping processes can be varied and studied.

In order to measure the properties of improved confinement and neo-classical transport very good and precise Te-, Ti- and Er- profile diagnostics are required
accompanied by accurate measurements of the power balance. This will be provided by Thomson scattering in both the core plasma and the divertor regions, supplemented by ECE measurements, interferometry / polarimetry, CX neutral particle analysis, CXRS spectroscopy, reflectometry and a fast Lithium beam diagnostic. In particular, interferometry can nicely be complemented by polarimetry (avoiding fringe losses during fast transitions) since in a stellarator the magnetic field is well known. The ports to implement a heavy ion beam probe (HIBP) in a later stage are provided. Its development is being considered and drift orbits are being investigated.

Confinement issues are addressed via:

- relative changes of pressure profiles
- confinement transitions
- effects of neo-classical transport due to changes in trapped particles (mirrors)
- effects due to magnetic field perturbations at rational iota values
- changes in the radial electric field
- local transport parameters from temporary evolution of injected test particles
- impurity accumulation and ECRH pump out
- radiation losses from plasma bulk, photosphere, islands and X-points

The different issues are covered by the diagnostics as given in table 2.

<table>
<thead>
<tr>
<th>Table 2</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Confinement and Transport</strong></td>
</tr>
<tr>
<td>local/non-local transport:</td>
</tr>
<tr>
<td>transient heat transport:</td>
</tr>
<tr>
<td>---------------------------</td>
</tr>
<tr>
<td>transient particle transport:</td>
</tr>
<tr>
<td>particle and impurity transport, anomalous transport:</td>
</tr>
<tr>
<td>pressure profile</td>
</tr>
<tr>
<td>ion energy balance</td>
</tr>
<tr>
<td>confinement transitions:</td>
</tr>
<tr>
<td>trapped particles, mirrors:</td>
</tr>
<tr>
<td>electron energy distribution, suprathermals:</td>
</tr>
<tr>
<td>magnetic field perturbations at rational values of iota:</td>
</tr>
<tr>
<td>radiation losses, bulk, corona, islands:</td>
</tr>
</tbody>
</table>
W7-X DIVERTOR

The properties of the magnetic configuration of W7-X with the formation of the "helical edge" and the associated diversion of magnetic field lines can be used as an inherent divertor. Two options for a divertor configuration, ergodic and island divertor, exist.

In case of the ergodic divertor the last closed magnetic surface is surrounded by an ergodic layer without large magnetic islands. Plasma wall interaction occurs close to the helical edges where field line diversion is a maximum. The target plates follow the helical edges and are appropriately shaped in order to minimize the local wall load ("helical trough").

The island divertor concept uses the existence of large islands at the boundary established for a magnetic configuration with moderate shear. Streaming along field lines the plasma passes the X-point region of the islands and arrives at the backside of the island after 4 to 5 transits. Whereas for the standard configuration with rotational transform close to 1 the 5/5-islands dominate the boundary, for varied values of rotational transform this role is played by the 5/6- or 5/4-islands.

The high complexity of the surface topology and its utilisation for island divertor operation requires sophisticated local measurement techniques to investigate edge island structures, particle densities and fluxes in the divertor chamber and onto the target plates, and heat load of the plates. Fairly good access to the divertor region for a whole set of diagnostics is possible. They are summarized in table 3.

Table 3
<table>
<thead>
<tr>
<th>Divertor Diagnostics</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>electron density:</td>
<td>Thomson scattering,</td>
</tr>
<tr>
<td></td>
<td>Langmuir probes, Li- and He-beams,</td>
</tr>
<tr>
<td></td>
<td>reflectometry, interferometry</td>
</tr>
<tr>
<td>electron temperature:</td>
<td>as before</td>
</tr>
<tr>
<td>ion temperature, impurity fluxes,</td>
<td>spectroscopy, Hα arrays, LIF,</td>
</tr>
<tr>
<td>drift velocities, neutral density:</td>
<td>press. gauges</td>
</tr>
<tr>
<td>thermal target load, power balance,</td>
<td>thermography, video, calorimetry</td>
</tr>
<tr>
<td>strike points:</td>
<td></td>
</tr>
<tr>
<td>radiation in X-point:</td>
<td>bolometry</td>
</tr>
</tbody>
</table>

STEADY STATE OPERATION

The inherent properties of a stellarator for steady state operation have implications on the diagnostics too. Although stationary operation is the ultimate goal, most of the experiments will be done with limited pulse duration of order 1 minute or 10 seconds in case of NBI. Different modes of operation are being discussed:

- pulsed discharges: purely ECRH heated, ECRH + NBI + ICRH at the ECRH resonant fields or starting with a non-resonant target + NBI beams + ICRH during B-field scans
- steady state ECRH operation for demonstration, special wall conditioning, specific studies of many different topics, possibly with slow changes of external parameters and magnetic settings
steady state ECRH with heavy duty NBI or ICRH pulses every 3 minutes.

These scenarios may not require a principally new diagnostic approach, however intelligent event recognition will be required and especially correlation studies will benefit from long integration times. Good spatial resolution will be possible with diagnostics which can be slowly scanned. Furthermore, temporally separated complex measurements by different diagnostics will be possible in the same volume. The new modes of operation mainly ask for a good feedback philosophy and flexible hardware to safely operate long pulse discharges with sequential operation of different subsystems to run through quite different plasma states and plasma parameters and to quickly respond to new ideas or needs.

PORTS AT W7-X

Despite the very complex coil set and 3D-geometry a huge number of quite large ports is available for heating and diagnostic access to the plasma. On the one hand the loss of toroidal symmetry and the 3D plasma structure, with changing shape in toroidal direction, complicate the diagnostics of the plasma. On the other hand, a five-fold symmetry still exists and the realization as linked mirrors provides nearly straight parts without curvature. Furthermore, due to the large aspect ratio of 10 and the absence of an ohmic transformer, inboard side access to the plasma is possible, which will especially be utilized for beam and divertor diagnostics. Additionally, the $\phi=18^\circ$ plane, with its drop like plasma cross section, allows for diagnostic access very far radially into the vessel, well suited for divertor diagnostics.
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6. M. Kick et al., "Overview of W7-X Diagnostics" Fusion Engineering and Design, accepted for publication 1996
Figure Captions:

Fig. 1: W7-X machine with the superconducting coil set, the vacuum vessel with ports and the cryostat mantle around.

Fig. 2: Development of one module showing the large number of ports also on the inboard side of the vessel.
PROBING $f_e(\vec{v})$ BY THOMSON SCATTERING ON W7-X

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Thomson scattering is a widely used standard diagnostic for $T_e, n_e$ measurements in fusion plasmas. Experience on W7-AS has shown, however, that during strong ECRH the scattering spectra can be distorted by superthermal electrons. In addition model calculations have shown that the distortion depends on the local magnetic field geometry. The Thomson diagnostic planned for W7-X provides for the first time sufficient access to the vacuum vessel to probe several scattering geometries, allowing for scattering vectors parallel, perpendicular and oblique with respect to the magnetic field. Furtheron a detection system with high spectral resolution is foreseen so that there will be a good chance to reconstruct the 2-dimensional electron distribution $f_e(v_\perp, v_\parallel)$.
MOTIVATION

The magnetic field of stellarators is modulated along flux lines by the helical field giving rise to local magnetic mirrors. Electrons in the long mean free path regime (LMFP) can be trapped in these mirrors causing loss cone effects. In addition, local electron cyclotron resonance heating (ECRH) can be responsible for distortions of the electron distribution function \[ S(\lambda) \]. A significant influence of these effects on the Thomson scattering distribution \( S(\lambda) \) has already been observed on the stellarator W7-AS.

Furtheron variations of \( S(\lambda) \) have been observed at the RTP Tokamak which have been connected with several dynamic effects \[ 2 \].

The future stellarator experiment W7-X has to demonstrate steady state operation. As Thomson scattering is foreseen as a standard diagnostic, it is necessary to provide additional information whether \( f_e(v) \) is Maxwellian or not.

EXPERIENCES FROM W7-AS

During electron cyclotron resonance heating with high power densities (highly focused wave launching) at low electron densities \( n_e \leq 6 \cdot 10^{19} m^{-3} \) a deviation between electron temperatures \( T_e \) measured by Thomson scattering and electron cyclotron emission (ECE) has been observed in the plasma centre (Fig.1).

The ECE data in this region depend mainly on the Maxwellian part of the distribution function. The deviations of the Thomson data are a consequence of interpreting experimental scattering spectra with superthermal tails by means of Maxwellian Thomson scattering functions (Fig.2). This interpretation fails even if the tail contribution to the scattering spectrum is outside the observed spectral range since the com-
plete behaviour of $S(\lambda)$ is changed.

The observed deviations depend on the toroidal position, on the angle of the scattering plane to the magnetic axis, and respond to variations of the magnetic field geometry (local extrema). Obviously, the Thomson scattering diagnostic is sensitive enough to detect variations of $f_e(\vec{v})$.

**TASKS FOR W7-X**

The task for Thomson scattering on W7-X is twofold:

1) Provide sufficient information to determine $f_e^{Max}(\nu)$ (for transport investigations, machine operation etc). Such a system will be located at a toroidal position not directly disturbed by heating or particle sources using a repetitive laser setup (Nd:YAG, Ti:Sapphire).

2) Investigate the feasibility of obtaining significant information on $f_e(r, \vec{v}) - f_e^{Max}(r, \nu)$. For this diagnostic a ruby laser scattering system will be employed with high spectral as well as high spatial resolution as developed at RTP [3]. This system will be located in a toroidal position where the ECRH is launched to obtain direct access to superthermal tails in a "magnetic mirror with heating". In addition, such a system improves the reliability of the standard diagnostic as deviations from the Maxwellian can be accounted for while determining $T_e$.

**SPECTRAL RANGE OF $S(\lambda)$**

The Thomson diagnostic on W7-X should be capable of measuring $T_e \approx 10\text{keV}$ in the plasma centre. In order to detect the influence of tail effects the spectral range should extend to $20..30\text{keV}$. For usual $\approx 90^\circ$ scattering, however, the corresponding spectral ranges are too large. More favourable conditions can be found at forward scattering angles due to the
(1 – \cos \Theta)^{-1} \text{ dependence of } S(\lambda).

**ANISOTROPY OF } f_e(\vec{v})

In order to study the behaviour of non-Maxwellian distributions in a magnetic field geometry with mirrors one needs access to several scattering vectors \( \vec{k} = \vec{k}_s - \vec{k}_i \).

1) Scattering vectors \( \vec{k} \parallel \vec{B} \) and \( \vec{k} \perp \vec{B} \) allow to study the behaviour of circulating vs. trapped electrons where the population of the latter is influenced by the ECRH as well as loss cone effects.

2) The region around the edge of the loss cone can be studied by oblique scattering vectors which is interesting especially in modular stellarators where the width of the loss cone is variable.

**FEASIBILITY**

The ruby laser scattering system will be capable of providing reasonable spectral and profile information in the ECRH launching region in the LMFP regime up to \( T_e \approx 5keV \) and for fixed scattering vector \( \Theta \approx 90^\circ \). For higher temperatures the spectral range can be restricted to a reasonable size if one uses scattering angles below 60° (Fig.3).

For a variation of the scattering vector the accessibility of the vacuum vessel becomes a big advantage. In order to probe \( f_e(\vec{v}) \) one has several possibilities (Tab.1) to combine scattering angles below 60° (to restrict the spectral range and increase the signals) with different scattering vectors (\( \vec{v} \) components).

As the distortions of the distribution function are most interesting around the magnetic plasma axis several small systems viewing mainly the plasma centre might be sufficient for probing \( f_e(\vec{v}) \).
We have made some preliminary studies to explore the sensitivity of such a system. As a measure of significance we have chosen the signal difference of two typical spectral channels:

\[ D(\varepsilon) = S(\varepsilon + \Delta \varepsilon) - S(\varepsilon - \Delta \varepsilon) \]

with the scattering function \( S(\varepsilon) \), the normalized wavelength difference \( \varepsilon = \Delta \lambda / \lambda_0 \) and the laser wavelength \( \lambda_0 \). The parameter \( \Delta \varepsilon \) corresponds to the half width of \( S(\varepsilon) \) i.e. the temperature. The resulting curves yield a significant signal difference as well as the most sensitive spectral range depending on the temperature and the scattering angle. The comparison of the positive and the negative branch (red and blue wings of the scattering function) allows to determine rather sensitively the influence of superthermal tails (Fig.4).

**CONCLUSION**

A Thomson scattering system to probe details of \( f_e(\bar{v}) \) seems to be feasible on W7-X making use of the good access through several ports. At present the port coordinates are being optimized to fulfill the requirements for the diagnostic in detail.

Some technical problems have to be solved, however. As long as high energy ruby lasers are used to provide sufficient high scattering signals the plane of polarization has to be matched to each scattering geometry.

The laser beam has to pass the scattering volume in several directions.
REFERENCES


CONCLUSION

A Thomson microscope was used to study plasma convection. It was found to be possible to observe the plasma convection patterns. However, the patterns were difficult to interpret. Further work is required to better understand the plasma convection patterns and to develop a model to explain the observed phenomena. The importance of these studies lies in the potential for improved understanding of plasma behavior in various plasma confinement systems. As the distortion of the distribution function around the plasma axis affects small systems viewing nearby, the plasma convection might be sufficient to explain these phenomena.
FIGURE CAPTIONS

Fig.1: $T_e$ as measured by Thomson scattering (full circles) and by ECE (open circles), where the latter mainly represents the Maxwellian bulk temperature \{ECRH(70GHz), P=400kW, focused launch, $B_0 = 2.5T$, $n_e(0) = 5 \cdot 10^{19}m^{-3}$\}. $r_{eff}$ corresponds to the magnetic coordinate.

Fig.2: Measured Thomson scattering distribution (symbols) for the same discharge. Upper curve: Fit of the relativistic Thomson scattering function which is used for the determination of $T_{Thoms}^e$. Lower curve: The same function with $T_{ECE}^e$ inserted (normalized to the same ordinate value). The difference between both curves is significant.

Fig.3: Left: Thomson scattering functions for $T_e = 1keV$ and $10keV$ (left) calculated from a Maxwellian and from a distribution function distorted by a small maximum at $v_\perp \approx 1.5 \cdot v_{th}$ (right). The scattering angle is 42°. The lower temperature represents a typical case for W7-AS, the higher temperature is about the maximum expected for W7-X.

Fig.4: Significance Test $D(\varepsilon)$ for several electron temperatures and scattering angles, calculated from a Maxwellian and a distorted distribution function as in Fig.3. For all temperatures an accessible scattering angle can be found where the complete scattering distribution fits into a reasonable small spectral range and the signal differences between the Maxwellian and the disturbed case are sufficiently large.

TABLE CAPTION

Table 1: Possible scattering angles and angles between $\vec{k}$ and $\vec{B}$ for three incident laser axes. Letters in column 'W7-X port' indicate ports of observation, 't,z,q' denote the ports of laser beam incidence.
<table>
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<th>$k_f^1$</th>
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<td>o</td>
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<td>136.6</td>
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W7-AS Contributions to the 12th International Conference on Plasma Surface Interactions
(20-24 May 1996, St.-Raphael, France)
A 3D Monte Carlo Code for Plasma Transport in Island Divertors

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Abstract

A fully 3D self-consistent Monte Carlo code EMC3 (Edge Monte Carlo 3D) for modelling the plasma transport in island divertors has been developed. In a first step, the code solves a simplified version of the 3D time-independent plasma fluid equations. Coupled to the neutral transport code EIRENE, the EMC3 code has been used to study the particle, energy and neutral transport in W7-AS island divertor configurations. First results are compared with data from different diagnostics (Langmuir probes, Hα cameras and thermography).

1. Introduction

Unlike in tokamaks, the magnetic structure and the discontinuous target plates in the W7-AS and W7-X stellarators introduce a three-dimensional plasma boundary. This implies that, for high recycling conditions, 3D effects of plasma, neutral and impurity transport have to be taken into account by both transport modelling and diagnostics [1]. In this paper, a fully 3D self-consistent Monte Carlo code EMC3 is presented, which solves, in a first step, a simplified version of the 3D time-independent fluid equations for plasma transport in island divertors. The main assumptions are $T_i=T_e$ (i.e. strong ion/electron collisional coupling), neglect of heat convection and parametrization of momentum losses due to viscosity and CX with neutrals. The particle, momentum and energy balance equations are solved iteratively in sequence. All diffusive terms are treated by following Monte Carlo particles in 3D magnetic coordinates. Particle diffusivity is assumed to be anomalous and the transport coefficients are taken from the
experiment. Energy transport is considered to be dominated by classical parallel and anomalous cross-field heat conduction. Islands, private flux region and targets plates are modelled in their real 3D geometry. High flexibility is provided by EMC3 to locally adjust the grid according to the required resolution, for example, finer grid near to the targets. Coupled to the neutral transport code EIRENE [2], the EMC3 code has been used to investigate the particle, energy and neutral transport in W7AS island divertor configurations.

The code has been first applied to the proposed island divertor of W7-AS, for which it predicts high recycling conditions associated with a strong density rise inside the islands. The results have motivated intensive experimental studies in order to clarify whether the predicted island high recycling can be obtained with the present inboard target plates. Simulations will be compared to data from different diagnostics.

2. Simplified one-fluid model

Although the transports of charged particles are best described by kinetic theory, solving the full kinetic equations remains an infeasible task. Instead, the edge plasma is commonly treated as a multi-fluid [3]. In the first step towards a full solution of the 3D Navier-Stokes equations, we assume a simple neutral plasma consisting of electrons and a single ion species. The plasma is considered to be dense enough that thermal equilibrium holds between electrons and ions, $T_e = T_i = T$. Parallel and perpendicular heat conductions are considered to dominate the energy transport. With a parametrization of the momentum losses due to viscosity and neutrals, the balance equations for mass, momentum and energy are given by

\[
\nabla \cdot (n\mathbf{v}_\parallel - D_\perp \nabla \rho) = S_p
\]

(2.1)

\[
\nabla_\parallel (m_i n \mathbf{v}_\parallel^2 + p) = \alpha_m
\]

(2.2)

\[
\nabla \cdot (-k_\parallel \nabla T - \chi_\perp n \nabla T) = S_e
\]

(2.3)

with the Bohm conditions for the particle flux $\Gamma_t$ and the heat flux $q_t$ at the target

\[
\Gamma_t = n C_t
\]

(2.4)

\[
q_t = \gamma \Gamma T \Gamma_t
\]

(2.5)
where $p$ is the plasma pressure, $S_p$ and $S_e$ are the volume sources of particle and energy determined by the EIRENE code and $\alpha_m$ is a constant momentum loss parameter determined by upstream and downstream measurements. The parallel energy conduction is considered to be classical, while the cross-field particle diffusion and energy conduction are assumed to be anomalous with $D_\perp$ and $\chi_\perp$ being determined from experiments. Particles are lost to the target plates at sound speed $C_s$ with energy $\gamma T$ where $\gamma$ is the energy transmission coefficient at the sheath. Plasma density and heat flux are given as boundary conditions at the separatrix.

3. Monte Carlo method

Due to the complicated geometry related to the island chain and the discontinuous target plates, the construction of a locally optimized grid for a finite difference solver may become problematic. This is not the case for the Monte Carlo method because of its high flexibility in the grid mesh construction and distribution. The grid can be locally adjusted according to the required resolution. The energy and particle transport equations are treated by the Monte Carlo method, whereas the momentum balance equation is solved analytically, as will be discussed later.

**Particle transport**

The particle transport equation (2.1) can be rewritten in the following time-independent Fokker-Planck form

$$
V_\parallel (-v_\parallel n) + V_\perp [\nabla_\perp (D_\perp n) - n \nabla_\perp D_\perp] + S_p = 0.
$$

Equation (3.1) describes the density distribution of particles diffusing across $B$ with the mean convection velocity $v_\parallel$. It can be solved by a random walk procedure [4]. Particles are randomly generated according to a volume or a surface source distribution on the LCFS, and then followed in small time steps $\Delta t$. The particle walks a step $\Delta r$ given by

$$
\Delta r = (4 D_\perp \Delta t)^{1/2} \xi_\perp + V_\perp D_\perp \Delta t + v_\parallel \Delta t
$$

(3.2)
where $\xi_\perp$ is the random unit vector perpendicular to B. Equation (3.2) indicates that the magnetic field line vector is required for tracing particles in two different directions. In order to avoid repeating the calculations of the magnetic field lines, it is convenient to trace particles in magnetic coordinates $(r^m, \theta^m, \phi^m)$, where the field line is a straight line with pitch $\Delta \theta^m/\Delta \phi^m = \xi_\perp$. Particles are traced and scored in the given grid mesh. If a particle is lost to a target plate, a new particle is generated and the computation continues until the desired accuracy of statistic is obtained.

In order to speed up the computation, it is necessary to represent the target plates in magnetic coordinates as well.

**Momentum balance**

In the absence of perpendicular coupling, the momentum balance equation (2.2) can be integrated immediately. Assuming that $v_\parallel$ reaches the sound speed at the target plate, we have simply

$$m_i n v_\parallel^2 + p = 2p_t - \alpha_m l_\parallel$$  \hspace{1cm} (3.3)

where $p_t$ and $l_\parallel$ are the thermal pressure at the target and the parallel distance from the target, respectively. Since the kinetic pressure, except in the region close to target plates, is a small quantity as compared to the thermal pressure, $v_\parallel$ can hardly be deduced directly from the equation (3.3) due to the statistical errors in $p$ resulting from the Monte Carlo method. Instead, we treat the problem in an indirect way. Note that $p = 2nT$ and $T$ remain unchanged during the iterative determination of $n$ and $v_\parallel$. From equation (3.3) we get

$$n_m = (2p_t - \alpha_m l_\parallel)/(m_i v_\parallel^2 + 2T)$$  \hspace{1cm} (3.4)

where the subscript $m$ on the density $n$ is introduced in order to distinguish it from $n$ as determined by the particle transport equation. Equation (3.4) gives a density distribution
required to satisfy the momentum balance. Comparing $n_m$ with $n$ and taking into account the fact that $n \propto 1/\nu$, we get $\nu_{\text{new}}$ for the next iteration

$$
\nu_{\text{new}} = (m_n n \nu^2 + 2nT)/(2p_e - \alpha_m 1_\parallel) \cdot \nu_\parallel.
$$

(3.5)

As seen from (3.5), $\nu_\parallel$ at the target plate remains unchanged. Therefore, an initial distribution for $\nu_\parallel$ satisfying the boundary condition $\nu_{\parallel} =$ sound speed at both target plates is required. This can be easily provided by assuming two groups of particles streaming initially antiparallel to each other. After replacing $\nu_\parallel$ in the particle transport equation (3.1) by $\nu_{\text{new}}$, the iteration starts and continues until $n$ and $\nu_\parallel$ converge.

**Energy transport**

With the definitions of $E = \langle n \rangle T$, $n_x = n/\langle n \rangle$ and $\kappa_{||} = \kappa_\parallel/\langle n \rangle$, where $\langle n \rangle$ is the average density at the separatrix, the energy transport equation (2.3) can be rewritten as

$$
\nabla_\parallel [\nu_\parallel (\kappa_{\perp} E) - E \nabla_\parallel \kappa_{\perp}] + \nabla_\perp [\nu_\perp (n_x \chi_{\perp} E) - E \nabla_\perp (n_x \chi_{\perp})] + S_e = 0
$$

(3.6)

where $S_e$ in the present version of the code, is the volume energy loss or gain due only to the hydrogen neutrals. The equation (3.6) is of the same form as the particle transport equation (3.1). Therefore, the procedure applied for the particle transport can be used for solving the equation (3.6). Now the Monte Carlo particles represent heat quantities. A subiteration process is required in this case, as the classical heat conductivity depends strongly on the temperature ($\kappa_{||} \propto T^{5/2}$).

**Solution sequence**

The computation starts with the energy transport, which is very sensitive to the temperature distribution. Particles representing heat quantities are randomly generated on the LCFS and followed initially in a source-free background plasma of constant density and temperature. The temperature distribution is then determined by iteration. Then, the neutral transport code EIRENE is called to provide the volume sources for the plasma. Particles started according to
$S_p$ from the EIRENE code are initially divided into two groups streaming antiparallel to each other in order to satisfy the sound speed boundary condition at both target plates. $v_{\|}$ is then corrected according to the momentum balance equation. Finally, $n$ and $v_{\|}$ are determined by sequential iteration of the particle transport and momentum balance equations. This step completes the sequence of the main iteration loop. After inserting $n$ and $S_e$ into the energy transport equation, the main and subiterations continue until $T$, $n$ and $v_{\|}$ converge.

4. Applications

The code was first applied to the proposed island divertor configuration of W7AS (Fig.1). The $t=5/9$ island chain is intersected by ten up/down symmetric target plates extending 18° toroidally. First results show a strong dependence of the ratio of parallel to perpendicular transport fluxes on the connection lengths governed by the island rotational transform $t_i$ [5]. It can be varied by changing the ratio modular/toroidal field currents and the current of the control coils. $t_i$ has to be sufficiently large to overcome the perpendicular transport across the islands, but sufficiently small to provide a sufficient radial power flow to allow ionization of the neutrals in the island core. For $t_i=0.1$, Figure 2 shows the density profiles across the island for three density values at the LCFS with the same heat flux through the separatrix ($q=0.5$ MW). Neither viscosity nor momentum loss via neutrals are taken into account ($\alpha_m=0$). In the high density case, the simulations predict a density rise by a factor two from the LCFS to the island O-point, indicating high recycling inside the islands. The temperature profiles for the three cases, on the other hand, do not significantly differ from each other, as the parallel heat conduction dominates the energy transport. The high density in the island can partially compensate for the relatively small radial distance from the target plates to the main plasma. Neutrals are ionized by more than 90% outside the LCFS and well confined by the baffles.

Recently, island divertor experiments at $t_a=5/9$ have been performed in W7AS with the islands intersected by 10 symmetric inboard target plates, aimed to explore the high recycling
conditions predicted by the theory. In this case, the simulations are concentrated on the density
dependence of the recycling behaviour, focussing on the effort to explain the experimental
observations.

Simulations are made for four NBI discharge series, covering a range of line-averaged density
\(<n_e>_{\text{line}}\) between \(2 \times 10^{19}\) and \(1.2 \times 10^{20}\) m\(^{-3}\). The heat fluxes across the inner separatrix are 0.4
MW for \(<n_e>_{\text{line}} = 2 \times 10^{19}, 4 \times 10^{19}\) and \(8 \times 10^{19}\) m\(^{-3}\), and 0.9 MW for the highest density, and
are estimated from the NBI power deposition and core radiation. Additional power losses in the
SOL are considered due only to the neutrals. The pressure drop factor \(\alpha_m\) is estimated from the
upstream and downstream measurements of two Langmuir probes [6]. The cross-field
diffusion coefficient \(D_\perp\) and the density at the inner separatrix \(n_{\text{sep}}\) are determined by the
Langmuir probe profile data across an island, and adjusted in such a way as to match the
measured profiles [6]. For \(<n_e>_{\text{line}} \leq 8 \times 10^{19}\) m\(^{-3}\), the resulting diffusion coefficient \(D_\perp\) is 0.6
m\(^2\)/s, dropping to 0.2 m\(^2\)/s as the density increases to \(1.2 \times 10^{20}\) m\(^{-3}\). In the last case, the
simulations show a density rise inside the islands (Figure 3), giving evidence of high
recycling. High density at the target leads to low temperature due to the power diversion along
the separatrix. The results for the four density values are summarized in Figure 4 which shows
the temperature and density profiles across the island from the inner separatrix to the target
plate. At relatively low densities \(<n_e>_{\text{line}} \leq 8 \times 10^{19}\) m\(^{-3}\), the density profiles in the island are
flat. High recycling occurs at \(<n_e>_{\text{line}} = 1.2 \times 10^{20}\) m\(^{-3}\), as indicated by a density rise by a factor
of 3 from the inner separatrix to the target. Temperatures near the plates decrease with
increasing density, but remain sufficiently high to allow ionizations of the neutrals in the island.
Strong pressure drop is experimentally found along the island fans, even for the low density
cases in which the neutrals could not make significant contributions. This is probably due to the
enhanced \(\perp\) anomalous viscosity related to the special island geometry in W7AS. The island
fans represent two close channels of antiparallel particle flows to the targets. Momentum loss
occurs if particles in the two channels exchange their momentum by means of cross-field
diffusion. This effect may be very pronounced in W7AS because of the small size of the islands.

Despite the small field line pitch in the island ($t_i=0.1$), the parallel heat conduction dominates the energy transport, which has been confirmed both by the thermography and the simulations. Figure 5 shows the energy flux distribution on a target plate resulting from EMC3 together with the surface temperature distribution from thermography. Both show a strong localization of the energy flux along the island fan with the largest connection length. In contrast to the energy flow which is governed by the island diversion, the particle flux depends more on the recycling conditions and the neutral flows. Figure 6 shows vertical scans of H$\alpha$ data from the diode array looking at the target and from the code. Both indicate that the recycling peaks at the outermost strike points. Simulations show that more than 60% of particles intersect the targets on the side, leading to a poloidally wide spread of the neutrals.

5. Conclusions

A 3D self-consistent Monte Carlo code EMC3 has been developed, aimed at investigating the edge transport in stellarators in the presence of islands. At present, the code solves a simplified version of the 3D time-independent plasma fluid equations. The code provides a high flexibility in the grid construction, so that it can be easily adapted to any complicated 3D edge topology of stellarators. Coupled to EIRENE, the EMC3 code simulates self-consistently the plasma and neutral transport in a full 3D space. Alternatively, the code can be used as a fit procedure to estimate the transport parameters by matching the local experimental data. In a first application to the proposed divertor configuration in W7-AS, the code predicts a density rise consistent with high recycling conditions in the islands. Similar behaviour has been observed experimentally in W7-AS with the inboard target configuration. The code results are supported by different diagnostics (Langmuir probes, H$\alpha$ diode array and thermography).
6. References

[1] F. Sardei, et al., this conference


Fig. 1: Arrangement of the proposed divertor for W7-AS.

Fig. 2: Density profiles across the island, averaged over the toroidal range of the plate. The three curves correspond to three densities at the LCFS: $5 \times 10^{18}$ (dashed), $1 \times 10^{19}$ (dot-dashed) and $2 \times 10^{19}$ (solid) m$^{-3}$.

Fig. 3: Contour plot of the plasma density.

Fig. 4: Density (a) and temperature (b) profiles across the island from inner separatrix to the target plate.

Fig. 5: Surface temperature distribution on a target plate from thermography (left) and the distribution of the energy flux from the EMC3 code (right).

Fig. 6: Vertical scans of H$_\alpha$ data from the diode array and from the EMC3 code.
Figure (a) shows the electron density profile, \( n_e / <n_e>_{sep} \), as a function of distance from the separatrix. The density values are indicated at specific points along the separatrix.

Figure (b) displays the electron temperature profile, \( T_e / eV \), with similar density values as in (a). The temperature decreases as the distance from the separatrix increases.
Experimental study on highly collisional edge plasmas in W7-AS island divertor configurations


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Abstract

Edge plasma scenarios in island divertor configurations („natural“ magnetic islands intersected by targets) are studied by comparing data from moderate to high density NBI discharges with 3D code (EMC3/EIRENE) results. The data strongly indicate that high recycling with significant particle flux enhancement was achieved in this geometry. But, plasma pressure losses towards the targets are relatively strong, and high recycling sets in only at \( \bar{n}_e > 10^{20} \text{ m}^{-3} \). The respective density enhancement in front of the targets is moderate (up to a factor of about three relative to the upstream density). These scenarios are also in basic agreement with B2/EIRENE code predictions. At \( \bar{n}_e \geq 1.5 \times 10^{20} \text{ m}^{-3} \) detachment seems to develop. Improvements are expected from additional loops controlling the field line pitch inside the islands, and from optimized targets which will better focus recycling neutrals into the islands. Both are in preparation.

1. Introduction

In optimized stellarators it is planned to utilize island divertors, which are based on inherent magnetic islands at the plasma boundary, for proper plasma exhaust. In the present W7-AS stellarator, crucial elements of this concept can be assessed; a full test will be carried out at W7-X. Like stellarators in general, W7-AS is strongly non-axisymmetric, and the edge topology is three-dimensional. The device can be operated with magnetic field configurations bounded by inherent, „natural“ islands of considerable size. The symmetry of the islands is \( 5\overline{5}m \) with 5 being the number of magnetic field periods and \( m = 7, 8, 9,... \) determined by the respective rotational transform \( \tau = n/m \) [1, 2]. The studies on W7-AS had to pre-clarify to what degree relevant boundary islands are stable with respect to equilibrium currents at finite plasma pressure \( \beta \), and were subsequently focused on the main questions whether high recycling and related divertor scenarios can be achieved with islands intersected by targets (open arrangement without baffles, see fig. 1a).
Section 2 describes the edge plasma experiments. In section 3, the specific edge topology chosen as optimum for this analysis is presented. In that context, results on the island stability at finite $\beta$ are briefly summarized. This issue is addressed in more detail in Refs. [3, 4]. Under these rather complex 3D conditions, the interpretation of local data from the experiments and characterization of respective edge regimes need sophisticated modelling adapted to the special island topology. Section 4 briefly summarizes the present state of code developments (for detailed descriptions see [5, 6]) and describes the adjustment of free input parameters of the utilized code to experimental data. In section 5, code results are compared with measured data, and section 6 gives a summary on the results and conclusions.

2. Experimental

The analysis was made for net current compensated NBI discharges at $B = 2.5$ T with ECRH start-up and balanced injection. Line-averaged densities $\bar{n}_e$ were varied between 2 $10^{19}$ and 1.5 $10^{20}$ m$^{-3}$. Heating powers were 0.8 MW for $\bar{n}_e \leq 8 \times 10^{19}$ m$^{-3}$ and 2 MW for higher densities. Data were taken during flat-top phases of about 300 ms (low to moderate densities) or 150 - 200 ms (highest densities). In the latter case, density control was lost in general after that time and the discharges were radiatively terminated before switching off the heating power. The edge rotational transform $t_e$ was 0.564 which corresponds to a configuration with 5/9 boundary islands. Edge plasma parameters were obtained from two Langmuir probes (CFC tips): a fast reciprocating probe (FRLP) at the position shown in fig. 1b, and a second probe close to a target. Both probes were operated in the single mode; $n_e$ and $T_e$ values were derived by fitting the probe characteristics up to the floating potential [7]. The measurements were completed by Thomson scattering, spectroscopic observation (H$\alpha$ diode arrays looking at the targets, CCD cameras with filters for H$\alpha$ or CIII radiation), bolometry, low-energy CX neutral analysis (LENA) and target thermography.

3. Edge topology

Magnetic field configurations with 5/m boundary islands calculated by the KW equilibrium code [8] show, up to $\beta_e = 1\%$, increasing island radial elongations and field line pitch, but intact island surfaces and preserved symmetry. This is in good agreement with experimental signatures from probe measurements, camera observations and thermographic patterns at the targets [2, 3] and holds, in particular, also for the configuration with $t_e = 0.564$ (5/9 islands) which was chosen as optimum for the present analysis. At $B = 2.5$ T, the $\beta_e$ range up to about 1% essentially covers the accessible discharge regimes studied. The 5/9 configuration offers both, sufficient main plasma cross section and relatively large boundary islands (figs. 1a, b). Typical island dimensions in the radial and poloidal directions are 5 - 10 cm (depending on the helical position). Field line connection lengths $L_c$ between stagnation points and targets for the probing line and relative probing positions $x/L_c$ (with $x$ being the parallel distance from the
stagnation point) are shown in fig. 2. As can be seen, moving along the probing line across the island does not only alter the perpendicular distance from the island separatrix, but also strongly $L_c$ and $x/L_c$. At higher $\beta$, $L_c$ becomes smaller (increased field line pitch, as mentioned above) and the island separatrix is shifted inwards, at the radial inside stronger than at the outside.

The target setup consists of ten poloidal graphite blocks (CFC) with toroidal widths of 12 cm, two per magnetic field period (one of them is shown in fig. 1, the second, covering the upper half of the radial inside, is at an equivalent, mirror-symmetric position). Effectively, they cover the inboard side of the plasma close to the toroidal midplane of each field period.

4. Modelling approaches

The edge plasma was modelled for the 5/9 island topology by a 2D (B2, time-resolving, multifluid [6, 9, 10]) and a 3D approach (EMC3 [4]), each coupled with the EIRENE neutral transport code [11, 12]. The B2 approach allows to study basic divertor properties with sophisticated physics, but includes helical averaging of the island configuration and can in particular not treat the present target geometry and hence the interaction with neutrals in a fully realistic way. Furthermore, it does not allow direct reference to local experimental data in the actual 3D configuration. The EMC3 code solves the set of fluid equations by a Monte Carlo technique considering the full non-axisymmetric geometry, but includes at present some physics simplifications: it treats single fluid plasma only, neglects heat convection and treats volumetric momentum losses by parametrization (see below). This is not caused by basic problems of the code, and an extension to more complete physics is under way. At present, we restrict ourselves to a „complementary“ description: the EMC3 code input parameters are adjusted to match local, experimental data, and the tendencies derived from this are then compared with B2/EIRENE predictions by brief reference in section 6.

The basic equations solved by the EMC3 code [4] are

\[ \nabla \cdot (n_{\parallel} v_{\perp} - D_{\perp} \nabla_{\perp} n) = S_p \]  
\[ \nabla_{\parallel} (m_{\parallel} n v_{\parallel}^2 + p) = -\alpha \]  
\[ \nabla \cdot (-\kappa_{\parallel} v_{\parallel} T - \chi_{\perp} n \nabla_{\perp} T) = S_e \]

with $n_e = n_i = n$, $T_e = T_i = T$, $p = 2nT$, $\kappa_{\parallel} \approx T_{ei}^2$ and the boundary conditions at the target $\Gamma = n_{ei} c_{ei}$, $q_{ei} = \Gamma_i \gamma T_i$. $n_e$, $n_i$ are the electron and ion densities, $v_{\parallel}$ is the parallel ion velocity, $m_{\parallel}$ the ion mass, $D_{\perp}$ and $\chi_{\perp}$ are the cross field particle diffusion coefficient and electron heat diffusivity, $T_e$ and $T_i$ the electron and ion temperature, $S_p$ and $S_e$ are volumetric particle and energy sources or sinks, the parameter $\alpha$ denotes momentum sinks, $\Gamma$ is the particle flux and $c_{\perp}$ the ion sound speed. $S_e$ considers only energy losses due to neutral hydrogen. The energy transmission factor $\gamma$ was set to eight [7].
The code has five free input parameters: $D_\perp$, $\chi_\perp$, $\alpha$, the power flow $P_*$ across the inner separatrix and the density $n_\alpha$ at the inner separatrix. $P_*$ was estimated from the deposited part of the NBI power and the total power radiated from the core (from bolometer data). $P_* = P_{\text{NBI, dep}} - P_{\text{rad,core}}$. In agreement with the density dependent relation of $\chi_\perp/D_\perp$ as observed in W7-AS at $t = 0.34$, $\chi_\perp$ was set to $3D_\perp$. The remaining free quantities $n_\alpha$, $D_\perp$ and $\alpha$ were obtained by adjusting them to match experimental densities from Langmuir probes at three points: at the position closest to the inner separatrix accessible by the FRLP without disturbing the core plasma (upstream position), a respective point within the private sector (see density profiles in fig. 3), and at the position of the second probe in the proximity of one of the targets ($x/L_c = 0.97$).

5. Results and discussion

Figure 3 shows $n_\alpha$ profiles across an island measured by the FRLP, and respective EMC3/EIRENE results. Island separatrix positions and the two points per profile, where the code results are matched to the experiments, are indicated. Data at the respective downstream matching points are shown in fig. 5. As can be seen from fig. 3, the measured profiles are, up to moderate line-averaged densities, rather flat inside the island and show a pronounced maximum close to the outer separatrix ("divertor tail") at $\bar{n}_e = 1.2 \times 10^{20}$ m$^{-3}$. At the highest density, the maximum of the profile is shifted inwards indicating (at least partial) detachment (see also Figs. 5 and 6). This latter interpretation is supported by camera observations of the H$_\alpha$ and CIII lines from the targets. The H$_\alpha$ stripes along the strike points (radial observation) vanish, the radiation becomes diffuse and much weaker. At the same time the CIII radiation (tangential view) becomes concentrated at the X-point proximity, and the integral power onto the targets is decreased to below 10% of the heating power (from thermography). Parallel with increasing density (loss of density control, section 2) the hot plasma cross section shrinks, and the discharges become terminated. The density profiles up to $\bar{n}_e = 1.2 \times 10^{20}$ m$^{-3}$ are quite well reproduced by the code (modelling of detached scenarios would not be reasonable with the present EMC3 version). Particle diffusion coefficients resulting from the code adaption are $D_\perp = 0.6$ m$^2$/s for $\bar{n}_e \leq 8 \times 10^{19}$ m$^{-3}$ and 0.2 m$^2$/s for $\bar{n}_e = 1.2 \times 10^{20}$ m$^{-3}$. The values of $D_\perp$ do, within factors of about 2, not critically affect the calculated profiles within the island, but are rather sensitive in reproducing the steep density drop towards the private sector. They are significantly smaller than derived for configurations with non-resonant boundary [13] and may indicate changed transport in island configurations, at least in the private sector. As is shown in fig. 4, the measured $T_e$ profiles are satisfactorily reproduced by the code only at low density. The discrepancy at higher density indicates stronger radiative losses due to impurities (mainly carbon) not considered by the code.

In figs. 5 and 6, respective upstream and downstream data from the experiments and the code are plotted versus the line-averaged density. As can be seen, the downstream density increases
up to a value of about three times the upstream density at $\bar{n}_x = 1.2 \times 10^{20}$ m$^{-3}$, whereas the downstream $T_e$ drops to about 10 eV. Energy losses due to neutral hydrogen reach up to about 50% of the power flow across the main plasma separatrix. Relative $H_\alpha$ intensities from the target show a similar tendency as the downstream densities and are well reproduced by the code. At high density, the scrape-off layer (SOL) becomes opaque for neutrals, and the total particle efflux from the SOL relative to that across the main plasma separatrix reaches up to a factor of about 35. Energy spectra of escaping CX neutrals from LENA (not shown) indicate, up to $\bar{n}_x = 8 \times 10^{19}$ m$^{-3}$, a linear decrease of the mean energy with increasing $\bar{n}_x$, but switch rather abruptly to much smaller mean energy at $\bar{n}_x = 1.2 \times 10^{20}$ m$^{-3}$. These are clear signatures for a high recycling scenario as it is known from tokamak divertors [7, 14]. At the highest density, the measured downstream density and $H_\alpha$ intensity show "roll over" indicating detachment as mentioned above. The present EMC3 version does not yet allow a quantitative appointment, but momentum transfer by charge exchange neutrals is, as in tokamak divertors, expected to be crucial for the transition from high recycling to detachment. Nevertheless, the data indicate strong plasma pressure losses along the island fans also in the attached regimes at low to moderate density which, in effect, shift the onset of high recycling to very high density and limit the respective density enhancement in front of the targets to moderate values. Though not fully comparable, B2/EIRENE results [6] show, at low to medium recycling, similarly excessive pressure gradients which are mainly balanced by the radial convective momentum flux and viscosity terms.

6. Summary and conclusions

The SOL of moderate to high density NBI discharges in a magnetic field configuration with "natural" boundary islands intersected by targets was analyzed with respect to divertor scenarios. Experimental data and EMC3/EIRENE code results strongly indicate that, different from limiter scenarios, a concentration of the particle sources inside the SOL and significant particle flux enhancement were achieved. But relatively strong plasma pressure drops towards the targets indicating considerable volumetric momentum losses restrict "classical" divertor high recycling scenarios to very high density ($\bar{n}_x \geq 10^{20}$ m$^{-3}$, $n_{er} \geq 3 \times 10^{19}$ m$^{-3}$) and limit the respective density enhancement in front of the targets to factors of about three relative to the upstream density. Quite similar scenarios are predicted by B2/EIRENE calculations [6] for the helically averaged island geometry and otherwise equivalent conditions, but high recycling sets already in at $n_{er} > 10^{19}$ m$^{-3}$ which is primarily appointed to the deviating target geometry in the 2D treatment. At $\bar{n}_x \geq 1.5 \times 10^{20}$ m$^{-3}$ the experimental data indicate detachment, but do not yet allow statements on the stability behaviour of this scenario. Further experiments on this issue at conditions with improved density control (NBI combined with 140 Ghz ECRH), together with
a code analysis considering also detailed momentum balance and light impurity radiation, are under way.

An extension of the density range with divertor high recycling and improved conditions for the establishment of stable detachment are expected from additional loops controlling the field line pitch inside the islands, and from helically more extended, optimized targets [3]. The latter will better approximate to 2D conditions and allow, in particular, improved focusing of recycling neutrals into the islands. Both additions are in preparation.

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Figure captions

Fig. 1. Cross sections of the W7-AS vacuum magnetic field configuration with 5/9 boundary islands at (a) a target plane and (b) a plane with the FRLP probing line.

Fig. 2. (a) Field line connection lengths $L_c$ between stagnation point and target as seen by the FRLP, and (b) relative, parallel distances $x/L_c$ of the probe from stagnation point for the vacuum magnetic field and at $\beta_o = 1\%$. Positions of the island separatrix are indicated.

Fig. 3. Electron density profiles measured by the FRLP across an island, Thomson scattering data measured close to the inner separatrix, and EMC3/EIRENE simulations of the probe data. Positions of the island vacuum field separatrix are indicated. In the discharges with high density and NBI power (see text), $\beta_o = 1\%$ is achieved which means that the actual separatrix positions are slightly shifted inwards. Code input data were adjusted to match the probe data at the innermost probe positions, at the outermost calculated points, and at a further probe position close to a target (see fig. 6).

Fig. 4. Electron temperatures $T_e$ from the FRLP and Thomson scattering, and respective EMC3/EIRENE simulations at (a) high and (b) low line-averaged density.

Fig. 5. Upstream and downstream $n_e$ and $T_e$ values (averaged along the strike line) from the EMC3/EIRENE code and data from the downstream probe versus the line-averaged density for the discharge series shown in fig. 3. Error bars indicate statistical errors, lines are for guiding the eyes only. Local downstream densities from the code were matched to the probe data.

Fig. 6. Power flow $P_x$ across the main plasma separatrix (input for the EMC3/EIRENE code, see text), total volumetric power losses $P_n$ due to neutral hydrogen, and particle flux enhancement factors $F_F$ from the code versus the line averaged density. Relative $H_x$ intensities from the target and respective values reproduced by the code (both normalized to unity at the lowest density).
Fig. 1a
A 2D Approach to Island Divertor Modelling for Wendelstein 7-AS

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Abstract

The properties of an island divertor for Wendelstein7-AS were investigated by using the B2-Eirene package. Because B2 is a two dimensional code the calculation needs to be done on a toroidally averaged grid. Carbon influx resulting from physical and chemical sputtering was treated self-consistently. The model predicts a transition from low to high recycling at a separatrix density of $1 \times 10^{19} \text{m}^{-3}$. A detailed discussion of the momentum removal mechanism for this geometry will be presented.

1. Introduction

The finite aspect ratio of a real stellarator leads to toroidal coupling between magnetic field components of different spatial symmetry. This toroidal coupling causes generally a destruction of the outer magnetic surfaces. Depending on the strength of the field components a chain of natural islands may emerge at resonant surfaces (resonance between the symmetry of the field component and the rotational transform). These natural islands can be used for divertor operation. The primary goals of a divertor are: to allow parallel energy losses to reduce the power load on the target plates, screening of the main plasma from impurities and provide high neutral gas density in front of the target plates to enable efficient pumping. To study the possibility of using the natural islands of W7-AS as a divertor configuration, a 2D-multi-fluid Code (B2) coupled with a 3D-Monte-Carlo neutral gas code (Eirene) was applied. In B2-Eirene practically all atomic physics relevant for describing edge plasma phenomena is incorporated. Therefore it can be used to determine leading parameters for an island divertor. Because B2 is a two dimensional code the calculation needs to be done on a toroidally averaged grid (section 2). In section 3 the main aspects of the B2-Eirene package related to this work will be introduced. The choices of the free parameters and boundary conditions will be explained in section 4. Mainly, two different sets of density scans have been made, one pure deuterium density scan and a multi-fluid scan (deuterium plus carbon) where carbon comes from physical and chemical sputtering from the target plates. In section 5 results from these calculations will be presented.

2. Grid generation for W7-AS

W7-AS is a modular stellarator with $n=5$ magnetic field periods. The value of the rotational transform $\tau$ can be chosen within the range of $0.2 < \tau < 0.7$. With increasing rotational transform the magnetic separatrix shrinks so that at the present limiter configuration, divertor action may be expected for values of $\tau > 0.4$. For this study a configuration with an edge rotational transform close to $5/9$ was chosen. In this case nine connected islands appear outside the main
plasma, as can be seen on the Poincaré-plot shown in Fig. 1. At present, ten symmetrically placed inboard limiters are installed in W7-AS that may be seen as target plates in this configuration. Using three different sets of Fourier coefficients for the inner, island and outer part of the plasma a three dimensional grid was constructed and made as orthogonal as possible. The geometry coefficients needed for the B2-calculation were averaged (distances) or summed up (areas, volumes) toroidally along the whole island. An averaged grid used for the neutral gas calculation shown in Fig. 2 was also constructed. The calculated region extends about four cm into the core region to cover the whole region where atomic processes involving neutrals are important.

3. B2-Eirene package

The B2(Braams)-Code [1] solves the continuity, parallel momentum and energy equations. Parallel transport is described by Navier-Stokes type equations. For radial transport an anomalous diffusive ansatz is made. The special topology in our case, see Fig. 2, is treated by three cuts in the numerical grid. Two cuts are needed to treat the periodic boundary condition at the connection to the neighboring island. The third cut provides a connection of the grid cells in the closed island (line from T3 to T4 in fig. 2). This cut technique originally developed for single-null tokamak divertor cases, where only two cuts are needed, could be straightforwardly extended to our special topology. For the parallel transport coefficients the classical values are used. The Eirene-Code [2] calculates from a given plasma state via a Monte-Carlo technique the neutral gas distribution and source terms for the B2 fluid model. Also physical and chemical sputtering is treated by Eirene to yield the carbon sources for the multi-fluid calculations. In that case the full 21-momentum approach is used [3]. All the calculations made in this study have been done by using the coupled B2-Eirene [4,5,6,7] package which also allows time-dependent calculations, however here we are only interested in stationary solutions.

4. Free parameters and Boundary conditions

The radial transport coefficients are assumed to be anomalous and their values have to be determined by comparison with experimental data. The radial diffusion coefficient was chosen according to the scaling law given in Ref.[8] which gives 0.5 m²/s for the higher density cases. For most calculations the same value (0.5 m²/s) was used for the radial viscosity and diffusion coefficient. This is justified by the assumption that turbulence is the reason for anomalous transport of both particles and momentum and this turbulence is approximately perpendicular to the magnetic field. Both values (radial viscosity and diffusion coefficient) were also varied in sensitivity studies. For carbon the same values as for deuterium were assumed. No pinch velocity was used for deuterium or carbon. Anomalous heat conduction for electrons was taken three times the particle diffusion coefficient (1.5 m²/s) and 1.0 m²/s for the ions. At the innermost flux surface of the grid the total power outflux from the core, deuterium density and zero net flux for all carbon ions was prescribed. This zero flux condition does not allow for the diffusion of lower charged carbon states into the core plasma, getting ionised and diffusing back into the calculation region. This results in a non-reliable CVI-CVII ratio close to the core plasma but should have only a minor affect on the lower ionised carbon states in the SOL or island region. The total power flux from the core plasma was set to 220 kW for the pure deuterium case (radiative losses in the core and edge plasma were taken into account) and 500 kW for the multi-fluid case (radiative losses from the core were taken into account). The density
was varied between $5 \times 10^{18}$-$5 \times 10^{19}$ m$^{-3}$. At the target plates sheath boundary conditions were used for all equations. Towards the private flux region, radial decay lengths (2 cm for densities and 3 cm for temperatures) have been fixed. At the center of the closed island (O-point) no net radial flux of particles, momentum or energy was assumed. Periodic boundary conditions have been used at the connection to the neighbouring island.

5. Results

The pure deuterium density scan shows a transition from low to high recycling at a separatrix density of about $1 \times 10^{19}$ m$^{-3}$. Radial density, electron temperature and pressure profiles at the midplane and along the target plate for two different densities are shown in Fig. 3, 4 and 5. The lower density case (circles in fig. 3) is a medium recycling case where the density drops along the separatrix from $7 \times 10^{18}$ m$^{-3}$ to $4.5 \times 10^{19}$ m$^{-3}$ at the target plate. In the case of a separatrix density of $1.3 \times 10^{19}$ m$^{-3}$ a high recycling regime characterised by an enhanced density at the target plate (black triangles fig. 3) compared to the midplane (open triangles) is found in the SOL. The calculated temperatures in front of the target plates are 50/25 eV (electron/ion) for the medium recycling case and 20/15 eV for the high recycling case. In both cases a strong radial dependence of the parallel pressure drop between the stagnation point and the target plate was found. The pressure drop varies from about 5 at the separatrix to about 2 just outside the closed island. Analysing the parallel momentum balance, keeping only the important terms for the cases just shown the pressure balance reads with: \( m \) ion mass, \( n \) density, \( u \) parallel velocity, \( v \) radial velocity, \( p \) total pressure, \( \eta_\perp \) radial viscosity, \( S_m \) momentum source from neutral gas calculation, \( x \) parallel, \( r \) radial coordinate:

\[
\frac{d}{dx} \left( mn u^2 + p \right) + \frac{d}{dr} \left( mn u v - \eta_\perp \frac{du}{dr} \right) = S_m
\]

(1)

Without any momentum loss a pressure drop by a factor of 2 results only from the acceleration to sound velocity at the entrance to the sheath of the target plate. To discriminate between convection and viscous momentum transfer the radial viscosity was reduced by a factor of 100 to 0.005 m$^2$/s. This reduction leads to a reduced pressure drop (about 4 at the separatrix) and an increased density enhancement in front of the target plates from 1.3 ($\eta_\perp = 0.5$ m$^2$/s) to 1.9 ($\eta_\perp = 0.005$ m$^2$/s). The momentum transfer can be seen from the radial profiles of the ram pressure (ratio of target plate/upstream sum of static and dynamic pressure) shown in fig. 6. For both values of $\eta_\perp$ we see an outward transport of momentum indicated by a momentum sink in the SOL (ram pr. < 1.) and a momentum source in the private flux region (ram pr. > 1.). The higher viscosity (open circles in fig. 6) leads to a radially broader distribution of the momentum sources. In this case ($\eta_\perp = 0.5$ m$^2$/s) the momentum losses from the eirene calculation contribute less than 10% to the pressure loss in the SOL. For the $\eta_\perp = 0.005$ m$^2$/s case the convective loss of momentum and loss due to neutrals are of the same size. The total integral pressure drop is about 10% for $\eta_\perp = 0.005$ m$^2$/s and 20% for $\eta_\perp = 0.5$ m$^2$/s with the following definition:

\[
1 - \left( \int_{\text{along target}} p^* d\Psi / \int_{\text{upstream}} p^* d\Psi \right)
\]

(2)

where $\Psi$ is the magnetic flux, and $p^*$ and the path of integration is explained in caption of fig. 6. Because the overall radial behaviour of the ram pressure in fig. 6 is nearly independent of $\eta_\perp$ we conclude that the radial momentum transport is mainly due to convective flux (see
equ. 1), which itself is driven by the strong gradients at the separatrix. Normally the value for the poloidal diffusion was chosen equal to the radial diffusion coefficient. A control calculation without poloidal diffusion gave high poloidal density gradients (factor 10 larger than radial gradient) at the outer stagnation point (line between T3 and T4 in fig. 2) which seems unrealistic high to us because of the strong turbulence in the SOL. That means poloidal diffusion must be considered in this geometry. The relative ratio between the strength of the parallel and cross-field transport depends on the field line pitch, which is in our case extremely small: B_0/B = 0.001-0.003 in the SOL. To be able to study this dependence of the field line pitch not only numerically, control coils will be installed at W7-AS, that enable modifications of the island rotational transform and size. The experimental transition from low to high recycling was found at a factor 3 higher separatrix density [9]. Reasons for finding a lower onset to high-recycling with this model may be caused by the 2 dimensional approximation which neglects the effect of the toroidally inhomogeneous plasma parameters, neutral gas and impurity distribution. Because of the problem measuring the total radiation level in our 3D-geometry the input power used in the calculation may differ from the one experimental available and this could explain the higher onset of the high-recycling regime.

The multi-fluid calculations also predict high recycling at separatrix densities larger than 1*10^{19} m^{-3}. For a separatrix density of 1.2*10^{19} m^{-3} a target density of 2.1*10^{19} m^{-3} and electron/ion temperatures of 18(18) eV are predicted. The total radiation level in this case was about 50% which is comparable to experimental data. Because the radiation layer is located within one cm in front of the target plates about half of the radiated power is also absorbed by the target plate, that only a power load reduction of about 25% is expected. About half of the power arriving at the target plate comes from the private zone. To get higher power load reductions a detached plasma state is necessary. Studies on the existence and the stability of a detached plasma in this geometry are under way. Clear shell structures for the different carbon species could be found (fig. 7). The CII-CIV layers are located just in front of the target, whereas CV is spread out through the whole SOL and island region. These results agree qualitatively with experimental results and may be used for quantitative verification via spectroscopic measurements.

6. Conclusion

B2-Eirene code was successfully used to model edge plasma behaviour on a toroidally averaged grid for W7-AS. Because of the small field line pitch cross-field transport plays an important role for the edge plasma parameters. High recycling with a density enhancement of about 1.5 and low temperatures in front of the target plates can be achieved in this configuration. In this case a parallel pressure drop of a factor of 5 was found at the separatrix which is balanced by convective radial momentum transport. Radial viscosity makes only small contributions to the momentum balance but still changes the plasma parameters close to the target plates up to 50%. To enable detailed profile comparison with experimental data for model-validation a full 3d-approach is necessary as it is explained in more detail in ref. [10,11].

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Fig. 1. Poincaré-plot of W7-AS for an edge rotational transform $t_e = 0.564$. The inboard limiter is shown shaded on the lower-left part of the figure.

Fig. 2. Geometrical grid resulting from toroidally averaging along one island. Radial profiles from point M1 to M4 are called midplane profiles. Profiles along the points T1, T2, T3, T4 are called target plate profiles. Notice that the part from T3 to T4 lies within the closed island.
Fig. 3. Radial density profiles at the midplane (open symbols) and along the target plate (full symbols) for a medium recycling and a high recycling case. The position of the profiles shown is explained in Fig. 2. The abscissa is centered at the separatrix position (point M2, T2 in Fig. 2), positive values are directed towards the center of the island. The points with abscissa values > 2.5 cm are located within the closed island region.

Fig. 4. Same as Fig. 3 except that the electron temperature is shown.
Fig.5. Same as Fig.3 except that the total pressure is shown.

Fig.6. Radial profile of the ratio $p^*_\text{target}/p^*_\text{up}$, where $p^* = p(1 + \gamma M^2)$, $p$ total pressure, $\gamma$ adiabatic coefficient, $M$ mach number. $p^*_\text{target}$ goes from T1 to T3 (fig. 2) and $p^*_\text{up}$ is taken along the line U1-U2-M4.
Fig. 7. Radially integrated (along the SOL-layer) parallel carbon density profiles. The partly overlapping $C^+$, $C^{2+}$, $C^{3+}$ layers can be seen.
Island Divertor Studies on W7-AS


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Abstract

Basic topological features of the island divertor concept for low shear stellarators are discussed with emphasis on the differences to tokamak divertors. Extensive measurements of the edge structures by two-dimensional plasma spectroscopy and by target calorimetry are in excellent agreement with predicted vacuum and equilibrium configurations, which are available up to central $\beta$ values of $\approx 1\%$. For this $\beta$ value the calculated field-line pitch inside the islands is twice that of the corresponding vacuum case. Video observations of the strike points indicate stability of the island structures for central $\beta$ values up to $\approx 3.7\%$. The interpretation of the complex island divertor physics of W7-AS has become possible by the development of the three-dimensional plasma transport code EMC3 (Edge Monte Carlo 3D), which has been coupled selfconsistently to the EIRENE neutral gas code. Analysis of high density NBI discharges gives strong indications of stable high recycling conditions for $n_e \gtrsim 10^{20} \text{ m}^{-3}$. The observations are reproduced by the EMC3/EIRENE code and supported by calculations with the B2/EIRENE code adapted to W7-AS. Improvement of recycling, pumping and target load distribution is expected from the new optimized target plates and baffles to be installed in W7-AS.

1. Introduction

In the past five years, increasing theoretical effort has been devoted to the exploration and optimization of suitable divertor concepts for low shear stellarators [1,2,3]. The interest in the island divertor approach is largely motivated by the favorable diversion
properties of the stellarator boundary in the presence of large "natural" magnetic islands (Section 2) and by the high geometric flexibility of such configurations. The island divertor concept can as well be applied to high shear helical systems like CHS and LHD. In CHS, a local island divertor (LID) has been installed in order to demonstrate its basic function and its efficiency in view of its future use in the LHD device [4]. Externally induced magnetic perturbations are also used to provide divertor action in the ergodic divertor of TORE SUPRA, which exploits a large stochastic boundary layer to reduce the target power load, extend the radiation zone and improve particle control and impurity screening [5].

The W7-AS "high ε" configurations, bounded by natural islands of considerable size, have the basic properties required for an island divertor. The target arrangement installed presently (see Section 4), is the first step on the way to an optimized divertor, which will start operation within the next two years. Fig. 1 shows the plasma column and the optimized target plates and baffles for the proposed W7-AS and W7-X divertors. Note the similarity of both configurations concerning aspect ratio and field periodicity as well as inherent threedimensionality of plasma and plasma facing structures. The present and the next step of the divertor development in W7-AS are intended to provide first experimental experience on divertor operation in a low-shear stellarator. In particular, they are aimed to clarify to what extent the vacuum edge structures are modified by plasma pressure, whether high recycling conditions similar to those of tokamak divertors can be achieved, and whether stellarator specific solutions will emerge for the exaust problem. Extrapolability to W7-X is not obvious, although some of the key edge parameters of W7-AS, as the radial island width and the power flux density across the separatrix, are comparable to those of W7-X. Owing to the complexity of the island topology and the strong toroidal inhomogeneity of the recycling processes, the development of three-dimensional transport models is a prerequisite for reliable prediction of the plasma behaviour and realistic interpretation of the experimental data.

After reviewing the basic features of the island divertor concept for low shear stellarators and the basic differences to tokamak divertors, this paper describes the experimental and modelling investigations on island divertor configurations of Wendelstein 7-AS during the past year. Section 2 introduces "natural islands" as inherent diverting structures of low shear stellarators, for which flux coordinates exist or can be derived, at least for all divertor-relevant configurations of W7-AS. Section 3 addresses basic properties of the island divertor related to the island topology and the toroidal inhomogeneities of plasma and targets and their consequences for plasma recycling and transport. A detailed comparison of measured and predicted edge structures for vacuum and moderate-β configurations is presented in Section 4. Section 5 briefly describes the models used
for transport investigations on W7-AS, namely the B2/EIRENE code adapted to the toroidally averaged island geometry of W7-AS and a new 3D code, EMC3/EIRENE, developed to simulate and interpret the experiments. In Section 6, results from a transport analysis of NBI discharges for different line-averaged densities are discussed and compared to simulations with the EMC3/EIRENE code. Section 7 gives a short outlook into the new optimized island divertor for W7-AS.

2. Natural islands in W7-AS

In island divertors of low shear stellarators as well as in tokamak divertors, the field lines outside the last closed flux surface (LCFS) are diverted to target plates. However, in the first case the diverted field lines form nested island surfaces surrounding completely the LCFS (Fig. 2). The plasma diffusing across the LCFS enters the island regions and is directed to the target plates located at the rear of the islands. The islands originate at resonances of sideband harmonics $B_{mn}$ of the B spectrum with the local values of the rotational transform $\tau = M_{\text{per}} m/n$ ($M_{\text{per}}$ = number of field periods). These harmonics reflect the inherent non-axisymmetry of the configuration and the properties of the modular coils. Typically, the largest islands occur at lowest-order resonances $m = 1$ ($\tau = 5/n$ for W7-AS and W7-X) and appear at the edge. Although shear increases towards the edge, there is generally only one low-order island chain governing the relevant boundary region. In W7-AS, the configuration with the 5/9 islands at the edge offers a good compromise between large plasma and large island size, and therefore has been chosen for standard divertor operation. Fig. 2 shows the islands in the "triangular" and "elliptical" cross sections (symmetry planes). Depending on the $\tau$ profile, the islands may be topologically closed or "open". In the first case (Fig. 2), standard flux coordinates are available for the islands and can be used for mapping experimental data and for defining a 3D modelling grid. If the edge $\tau$ value is slightly decreased with respect to this case, the island chain is shifted outside and the island flux surfaces eventually break up beginning at the separatrix (open island case). What is left finally is a chain of strongly diverted island fragments, regularly distributed according to the poloidal mode number of the island resonance. These regular structures can be represented by nested "open magnetic surfaces", spanned by field lines, for which generalized flux coordinates can be obtained [6]. This procedure provides a mapping framework for all open edge configurations of W7-AS in the $\tau$ range relevant for divertor operation.

3. Island divertor vs. tokamak divertor

Unlike in tokamak divertors, the X-lines in island divertors are helical, with the pitch
given by the resonant $\varepsilon$ of the island chain. The poloidal progression of the field lines in the island reference frame is a measure of the internal rotational transform inside the island, $\varepsilon_I$, and determines the connection length, $L_c$, from target to target. As in tokamaks, the SOL extends poloidally from the stagnation point via X-point down to the targets (Fig. 3). However, the island SOL is poloidally closed in front of the targets, which enables trapping of recycling particles. The short distance between the targets and the LCFS ($\approx 5$ cm for W7-AS and $\approx 8$ cm for W7-X, standard configuration), requires higher plasma densities than in comparable tokamaks to effectively decouple the recycling neutrals and the target-released impurities from the plasma core. On the other hand, recycling inside the island tends to raise the local density and hence to improve the screening of neutrals and impurities. This effect is illustrated schematically in Fig. 4. If the main plasma density is small (upper drawing), ionization takes place in the main plasma and the density drops from the separatrix into the island (island low recycling). Increasing the plasma density shifts the ionization into the island (lower drawing), and the trapped particles lead to a density rise inside the island (island high recycling).

The efficiency of this process depends strongly on the distance of the targets from the LCFS and on $L_c$, i.e. on the radial position, radial width and internal field-line pitch of the islands. These quantities can be optimized by varying $\varepsilon$, the vertical field or the currents of control loops [7], which change the radial and poloidal field components. $L_c$ has a crucial impact on the divertor performance, as it directly affects the balance between parallel and radial transport. It has to be large enough to provide a large parallel temperature gradient and a sufficient power flux into the islands to sustain the recycling process there, but small enough to prevent a substantial cross-field transport to the targets. Relevant $L_c$ values are 100 to 300 m for W7-AS and W7-X.

The poloidal width of the islands, i.e. the distance between the two divertor fans, scales roughly as $a/n$ ($a =$ radius of the LCFS), giving $\approx 5$ cm (in the radial average) for the 5/9 resonance of W7-AS, with $a =$15 cm and $n = 9$. Since the two fans carry particles flowing in opposite directions, this small distance may induce strong (anomalous) radial shear viscosity inside the islands, contributing significantly to the total parallel momentum losses. This effect is not known in tokamaks.

A second basic geometric difference is the intrinsic three-dimensionality of island divertors. Besides the toroidal variation of the island shape, which leads to higher radial coupling of the flux tubes at positions of smaller radial width, a parallel modulation of the plasma parameters is introduced by the discontinuous target plates [8,9]. This is due to the strong periodic interaction of the outflowing plasma with the recycling neutrals and target-released impurities. (Note that the total wetted target
length is $\approx 1/18$ of the total island length for W7-AS (optimized divertor) and $\approx 1/10$ for W7-X (Fig. 1.) For high recycling conditions associated with low temperatures $T \leq 10$ eV in the downstream region, parallel heat conduction is too low to remove the local gradients of density and temperature arising from plasma/neutral interaction at the toroidal positions of the target plates. These modulations are superimposed to the standard parallel gradients associated with the poloidal diversion of the flux tubes towards the target plates. They require an adequate toroidal resolution of diagnostics and modelling of the downstream plasma. Furthermore, discontinuous target plates must be tilted toroidally with respect to the field lines in order to avoid leading edges arising from the poloidal progression of the field lines, poloidal diffusion or poloidal drift motion. The tilt determines the target load density and the toroidal length of the target plates.

Both the peaking of the density inside the islands for high recycling conditions and the parallel modulation of the plasma parameter profiles due to discontinuous target plates have been predicted by the new 3D Monte Carlo edge transport code EMC3 (see Section 5), applied to the optimized W7-AS target plates (see Section 7) [8,9].

4. Edge topology

As a first step towards an island divertor for W7-AS, the previous two up/down rail limiters have been replaced by a set of 10 inboard target plates (segmented CFC graphite blocks of 23 cm poloidal and 12 cm toroidal size each). The plates are placed toroidally on both sides of the triangular cross sections (Fig. 5) and preserve both the inherent fivefold periodicity and up/down symmetry of the configuration. Both limiter and divertor operation are possible, depending on whether the target plates intersect regular flux surfaces or islands (Fig. 5). For divertor operation with a given edge resonance $n_0 = 5/n$ (Fig. 5), the inboard position of the targets allows an easy variation of the connection lengths by application of a vertical field shifting the configuration horizontally with respect to the targets.

Extensive experimental investigations have been carried out in order to check the reliability of the predicted island structures and their geometric integrity against $\beta$ effects. These are prerequisites for a successful island divertor operation and for a realistic use of transport models based on predicted topology. Fig. 6 compares measured and calculated edge structures for a $n_0 \approx 5/8$ configuration at low $\beta$ (central $\beta_0 = 0.25 \%$). The video image (right picture) is a tangential view through $\approx 2.5$ m of the plasma column in the light of carbon (CIII, 467 nm). With an ionization potential of 48 eV, this species resides in a small radial shell close to the LCFS. The contours shown in the left hand side of Fig. 6 visualize successive poloidal cross sections of a
single vacuum flux surface next to the LCFS in a perspective projection as seen by the video camera. The most prominent cross section at $\phi = 17^\circ$ is emphasized by a bold contour. The central picture results from the perspective projection of the flux surface itself, weighted with a constant density distribution. All relevant features of the video picture, including the multiple structures at the bottom and at the inboard (left) side, are reproduced by the simulations. The excellent agreement can be regarded as a three-dimensional verification of the edge structures close to the inner separatrix of the 5/8 vacuum island chain.

Experimental evidence of the edge structures close to the outer separatrix of the 5/8...5/11 vacuum island chains is provided by the flux of the recycling particles ($H_\alpha$ light) and by the heat load (target calorimetry) across the 8 tiles of a target plate (Fig. 7). The islands are well defined and closed, the target plates have limiter function and no divertor action is expected in these cases. The maxima of the profiles clearly appear at the intersections of the corrugated island boundary with the target. The minima reflect the usual radial decay of the plasma parameters in a limiter SOL. The phase shift of the island position between resonances with even and odd poloidal mode numbers is well reproduced by the experimental profiles. The small amplitude factor of the $H_\alpha$ profile modulation with respect to that of the heat load is mainly due to a) the temperature decay in the limiter SOL, b) the deviation of the ionization to the $H_\alpha$ emission rate at low temperatures and c) the poloidal spread of the neutral fluxes.

If, however, the islands are intersected by the targets, divertor fans are formed along the outer island separatrices. The island chains shown in the Poincaré plots of Fig. 8 refer to vacuum and $\beta_0 = 1 \%$ conditions, respectively, with the same external currents, corresponding to $\vare_\alpha \approx 5/9$ configurations. The $\beta_0 = 1 \%$ configuration was obtained with the KW equilibrium code [10]. (Higher-$\beta$ equilibrium calculations are not available at present.) It shows an increased radial elongation of the islands due to field modifications from the internal plasma currents. The most relevant change, however, is the doubling of the field-line pitch inside the islands, which reduces the connection length to half of its value. The island surfaces remain intact and their phases unchanged. This is confirmed by the video pictures of Fig. 8, showing the island strike points via $H_\alpha$ light emission of recycling particles. (The 8 tiles of the target are visualized in an overlay to the video picture.) For both configurations, the locations of the observed $H_\alpha$ stripes on the target coincide very accurately with the predicted strike points shown on the right, proving the reliability of the calculated equilibrium edge configuration. $\beta_0 = 1 \%$ is slightly below the highest $\beta$ value of the discharges analyzed in the present study (Section 6). However, the structure and phase of the observed poloidal stripe sequence are found to be stable up to $\beta_0 \approx 3.7 \%$, corresponding to $< \beta > \approx 1.8 \%$. This demonstrates the robustness of the 5/9 islands
5. Transport models

Sophisticated 2D plasma transport codes, coupled with Monte Carlo codes for the neutral gas, are widely used in tokamak divertors to interpret experimental data, improve the basic understanding of plasma and impurity behaviour and predict and optimize the performance of new divertors [11,12,13]. In island divertors, additional effects introduced by the island topology and the toroidal target discretization (see Section 3) complicate the description of the divertor physics compared to tokamaks, motivating the urgent need of adequate 3D models. In the present study, the plasma edge transport of W7-AS was modelled with the B2 code [14,15] and the EMC3 (Edge Monte Carlo 3D) code, which was developed recently for island divertors [8,9]. Both codes are coupled with the EIRENE code [16] for the neutral transport. The chosen 5/9 edge configuration is described by three distinct sets of Fourier coefficients representing the island chain and the adjacent flux surfaces at the radial inside and outside of the islands (Fig. 9b). The islands are intersected by the targets outside the O-points. In the B2 code, a 2D grid is obtained by helical averaging of islands and targets (over 9 field periods), which results in a single island configuration, bounded by the target, two private flux regions and the main plasma (Fig. 9a). The code requires an orthogonal grid, which can be realized only approximately in W7-AS. Even with these restrictions, the B2/EIRENE code can be used to estimate basic effects of the island geometry on the divertor physics in all regimes and to provide 2D reference solutions and benchmarks for the EMC3 code.

The EMC3 code is a fully selfconsistent 3D Monte Carlo code which solves, in a first step, a simplified version of the 3D time-independent plasma fluid equations. The main assumptions of the present first version of the code are single fluid plasma, neglect of heat convection and parametrization of momentum losses from experimental data. All diffusive terms are treated by following Monte Carlo particles in magnetic coordinates. Islands, private flux region and target plates are modelled in their real 3D geometry (see Fig. 9b, showing the grid at the cross section of the targets). High flexibility for resolving strongly diverted magnetic structures and high gradients near the targets is provided by a locally (3D) adjustable, non-orthogonal grid. The 3D particle, momentum and energy transport equations, coupled with the EIRENE code for the neutral gas, are solved iteratively in sequence. Particle and energy sources from EIRENE code are included in the transport equations. Standard Bohm conditions are assumed at the target plates. $D_{x,\perp}, n_{e,sep}$ (plasma density at the LCFS), $P_{sep}$ (power flux across the LCFS), $\alpha$ (total parallel momentum loss parameter) are input quantities
obtained from experimental data. In the present study, $P_{sep}$ was taken from the NBI power deposition after subtracting the plasma core radiation (estimated from Soft-X and bolometer data) and $\chi_{\perp}$ was set equal $3*D$, according to previous estimations from density scans at $\tau = 0.34$. The remaining three parameters were chosen so as to match Langmuir probe data at upstream and downstream positions [17]. Parallelization of the EMC3/EIRENE code is under way.

Since B2 describes sophisticated physics in a simplified (2D) geometry, whereas EMC3 describes simplified physics in a sophisticated (3D) geometry, the two codes are of complementary use for island divertor investigations on W7-AS.

6. Transport results

The plasma transport analysis presented here refers to net-current compensated NBI discharges at $B = 2.5\ T$ with ECRH start-up and balanced injection. The configuration was defined by the edge iota parameter $\iota_{\alpha} = 0.564$, which corresponds to a 5/9 island chain intersected by the targets outside the O-point (see Section 5). Line-averaged densities $\bar{n}_e$ were varied between 0.2 and $1.5 \times 10^{20}\ m^{-3}$. Heating powers were 0.8 MW for $\bar{n}_e \leq 0.8 \times 10^{20}\ m^{-3}$ and 2 MW for $\bar{n}_e = 1.2 \times 10^{20}\ m^{-3}$. A central $\beta$ value of $\approx 1.2\ %$ was estimated for the 2 MW discharges. The data for the analysis were obtained during flat top phases of about 300 ms (low to moderate densities) or 150-200 ms (highest densities). In the latter case, density control was lost after that time and the discharges were terminated softly by radiative collapse. $n_e$ and $T_e$ data were provided by two Langmuir probes: a fast-reciprocating probe (FRLP) crossing an island at a position far from the target (Fig. 10) and a second probe close to the target. The measurements were supplemented by Thomson scattering, spectroscopy ($H_{\alpha}$ diode arrays looking at the targets, CCD cameras for $H_{\alpha}$ and CIII radiation), bolometry, low energy CX neutral analysis (LENA) and target thermography.

The results for the density range mentioned above are summarized in Fig. 11 and Fig. 12. Fig. 11a shows density profiles along the path of the FRLP across the island (Fig. 10), and the respective results from the EMC3/EIRENE code. Indicated in the picture are also the island-bounding inner and outer separatrices and the two points per profile, where the code results are matched to the experiments. (A third matching point is located at the downstream probe close to the target (see [17] for details)). Thomson scattering data are also shown for reference. The measured density profiles are rather flat except for the highest $\bar{n}_e$, for which the density peaks close to the outer separatrix. In all cases the profiles are well reproduced by the code. For $\bar{n}_e \leq 8 \times 10^{19}\ m^{-3}$, the diffusion coefficient resulting from the modelling (after adjusting the input parameters to the experiment) is $0.6\ m^2/s$, dropping to $0.2\ m^2/s$ for the highest density. These
values are smaller than those resulting from the scaling derived for limiter configuration without boundary islands [18].

The corresponding measured and simulated temperature profiles for the highest and lowest \( \bar{n}_e \) are shown in Fig. 11b. The higher temperatures predicted for the first case may be ascribed to the missing impurity radiation in the code.

Fig. 12 shows upstream and downstream densities and temperatures for the same discharges as functions of \( \bar{n}_e \). At \( \bar{n}_e \approx 10^{20} \), abrupt steepening of the downstream density (up to three times the upstream value), of the flux amplification factor (up to 35) and of the measured and calculated \( H_\alpha \) is shown, whereas the downstream temperature drops to about 10 eV and the total neutral power losses rise to \( \approx 50 \% \) of the power crossing the LCFS. Furthermore, the CX spectra of the escaping neutrals from LENA (not shown here) shift to much smaller mean energies. All these results together give experimental evidence of high recycling conditions for \( \bar{n}_e \geq 10^{20} \text{ m}^{-3} \). At the highest densities shown in Fig. 12, \( \bar{n}_e = 1.5 \times 10^{20} \text{ m}^{-3} \), the measured downstream density and \( H_\alpha \) drop again, indicating detachment. This interpretation is supported by several additional observations from the plasma boundary [17]:

- the maximum of the Langmuir probe density profile is shifted inwards;
- the CIII radiation from the targets (tangential video view) is shifted towards the X-point, indicating a shrinking of the plasma column;
- the \( H_\alpha \) radiation at the targets (radial video view) is strongly reduced and its modulation across the targets (\( H_\alpha \) stripes at the strike points, see Section 4) disappears;
- the total target load (from thermography) decreases below 10 % of the heating power.

However, density control was generally lost at that time and the discharges terminated in a radiative collapse. Experiments with improved density control (by combined NBI and ECR heating) are planned in order to assess the feasibility of a stable detached plasma regime for both the present and optimized divertor arrangements.

Fig. 12 also shows a significant pressure drop even for low densities, for which momentum loss by CX with neutrals cannot contribute significantly. This is probably due to enhanced radial shear viscosity related to the small island size of W7-AS (see Section 3). The 2D distribution of density and temperature in the triangular cross section, obtained with the EMC3/EIRENE code for the \( \bar{n}_e = 1.2 \times 10^{20} \text{ m}^{-3} \) case, is visualized in Fig. 13.

The thermal load on the target plates, as resulting from target thermography and
EMC3/EIRENE heat flux calculations, is not homogeneous (Fig. 14), indicating strong localization of the heat fluxes along the island fan with the largest connection length, which collects a large power fraction crossing the LCFS. This island fan corresponds to the strike point located at largest poloidal distance from the midplane (see, for example, bottom strike point in Fig. 8). The connection lengths to the other strike points is much smaller due to mutual shading of the upper and lower target plates.

7. New optimized divertor for W7-AS

Improved divertor performance in W7-AS is expected from the planned additional control coils and optimized target plates and baffles [2]. Very small field perturbations ($\approx 10^{-4}$ of the main field) introduced by the additional current loops will be sufficient to modify significantly the island size and position, as well as the connection length. This will increase the flexibility of divertor operation. The new 10 target plates (Fig. 1) will be placed symmetrically at top and bottom of the elliptical cross sections (Fig. 15), where the islands have their largest radial extent and their largest distance from the magnetic axis. Homogeneously wetted target areas and a careful avoidance of leading edges will provide recycling fluxes focussing predominantly into the islands. For the new divertor, the EMC3/EIRENE code calculations predict the onset of high recycling at lower densities than for the present target arrangement. Furthermore, neutral compression due to target inclination and baffles leads to neutral densities in the divertor chamber more than two orders of magnitude higher than outside of it, in contrast to the poloidally widespread distribution of the neutrals predicted for the present targets (Fig. 16).

8. Summary and conclusions

Different from tokamak divertors, the edge topology of island divertors in low shear stellarators is intrinsically three-dimensional. Edge resonance, island size and position and connection length are variable quantities, which can be optimized by varying $\varepsilon$, $B_z$ and the currents of control coils. Discontinuous target plates imply a toroidal localization of recycling, which leads to locally higher densities and improved neutral screening from the main plasma, thus compensating, to a certain extent, for the shorter distance between the target plates and the main plasma. Strong radial shear viscosity contributing to significant parallel momentum losses may arise by friction between particles flowing in opposite direction along the island fans, which are very close to each other in W7-AS. The target load is generally two-dimensional, and diagnostics and modelling of the low temperature downstream plasma need adequate parallel resolution.
The vacuum edge corrugations due to the 5/n resonances are reflected with high accuracy in tangential video pictures and particle and energy deposition profiles over the target plates. For islands intersected by the target plates, equilibrium calculations up to $\beta_0 = 1\%$ accurately reproduce the strike points of the island fans, as visualized by $H_\alpha$ (video) profiles over a target plate. The equilibrium currents increase the field-line pitch inside the islands, which, for $\beta_0 = 1\%$, becomes twice as large as that of the vacuum field. This implies shorter connection lengths, which have to be taken into account in the transport modelling. The observed $H_\alpha$ stripe sequences are stable over a central $\beta$ range up to $\beta_0 \approx 3.7\%$.

3D transport modelling has been shown to be essential for the interpretation of island divertor experiments. The new EMC3 transport code, coupled selfconsistently with the EIRENE code, has been applied to the interpretation of a high recycling 3D island divertor plasma. In the present version of the code, the parallel momentum losses are not yet included explicitly, but they are parametrized from experimental data.

Experimental evidence of high recycling conditions for $n_e \geq 10^{20}$ m$^{-3}$ and 1 MW power across the separatrix is indicated by peaking of the density inside the islands and close to the targets as well as by steepening of target-$H_\alpha$ emission as function of $n_e$. For $n_e = 1.2 \times 10^{20}$ m$^{-3}$, the flux amplification factor becomes 35 and the downstream density rises to three times that of the upstream value, as deduced from the EMC3/EIRENE code simulations. The parallel pressure drop is relatively high even for low line-averaged densities, indicating a significant contribution from radial shear viscosity. For highest line-averaged densities of $1.5 \times 10^{20}$ m$^{-3}$, the downstream density decreases again, suggesting rollover consistent with detachment. The probe density profiles and the $H_\alpha$ profiles on the targets are reproduced satisfactorily by the EMC3/EIRENE code simulations. For the same discharges the B2/EIRENE code, adapted to W7-AS by helically averaging islands and targets, indicates onset of high recycling at lower upstream densities, which is probably due to the assumed toroidal and poloidal symmetry of the idealized targets in the code.

The new W7-AS divertor will be optimized with respect to target position, load distribution, recycling and pumping efficiency. Compared to the present targets, the EMC3 code predicts, for the new divertor, a stronger neutral compression near the targets. Ionization will be focussed inside the islands, leading to higher densities there and at the target plates for the same upstream density and input power. This also implies a transition from low to high recycling at lower upstream densities.

For W7-X and larger stellarators, the increasing size of the islands will reduce the plasma flux to the targets due to cross-field transport, the penetration of the neutrals
into the plasma core and the momentum transfer across the islands by radial shear viscosity. That is, within the toroidal range of the targets the island divertor physics will become closer to 2D tokamak-like conditions, except for the confining effects of closed islands. However, all positive and negative implications of toroidal plasma inhomogeneity and discontinuous targets will remain.
References


[8] Y. Feng et al., this conference.


Figure captions

Fig. 1: Plasma column and divertor target and baffle arrangement of W7-AS ($R = 2\,\text{m}$) and W7-X ($R = 5.5\,\text{m}$). Note the discontinuous divertor elements compared to tokamak divertors.

Fig. 2: "Natural" island chain of the $\epsilon = 5/9$ resonance in the "triangular" and "elliptical" cross sections of W7-AS.

Fig. 3: Upstream and downstream SOL of a magnetic island intersected by a target plate.

Fig. 4: Principle of low and high recycling inside islands intersected by target plates.

Fig. 5: Present target plate arrangement in W7-AS.

Fig. 6: Measured (video image of CIII, right picture) and calculated (left and central pictures) W7-AS edge structures for a $\epsilon = 5/8$ resonance. The calculated contour and density plots are perspective views of a single flux surface close to the LCFS.

Fig. 7: Particle flux ($H_\alpha$) and energy deposition (target calorimetry) profiles on a target plate for different edge resonances. The profiles peak at the intersections of the corrugated island boundary with the target.

Fig. 8: $H_\alpha$ view (video) of the strike points of the 5/9 island fans for low and moderate beta values (left) and corresponding Poincaré plots of vacuum and equilibrium calculations (right). The island surfaces remain intact and the strike points stable against $\beta$ effects.

Fig. 9: 2D grid used in the B2 code and 3D grid used in the EMC3 code for a 5/9 island divertor configuration. The 3D grid is shown at the cross section of the targets.

Fig. 10: Path across an island of the fast-reciprocating Langmuir probe used for comparison of measured and calculated density profiles.

Fig. 11a: Measured and calculated density profiles along the probing path shown in Fig. 10 and Thomson scattering data.

Fig. 11b: Temperature profiles for the highest and lowest line-averaged densities shown in Fig. 11a.
Fig. 12: Upstream and downstream densities, neutral power loss, flux amplification factor and $H_\alpha$ from experimental and code data as function of the line-averaged density. Rollover of density and $H_\alpha$ at the highest densities indicate detachment.

Fig. 13: Two-dimensional plot of the calculated density and temperature for the high density case shown in Fig. 11.

Fig. 14: Thermal load distribution over the target plates from thermography (left) and 3D heat flux calculations (right).

Fig. 15: Cross section of the optimized island divertor arrangement for W7-AS (see also Fig. 1).

Fig. 16: Typical two-dimensional neutral density distributions in the poloidal plane of the present (left) and optimized (right) target plates for W7-AS.
W7AS

W7-X

Fig. 1
23rd European Physical Society Conference on Controlled Fusion and Plasma Physics
(24-28 June 1996, Kiev, Ukraine)
Radiative boundary studies in the Wendelstein 7-AS stellarator


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1. Introduction

Radiative edge cooling is widely considered necessary for power exhaust of future steady state fusion devices in order to prevent target overload by high heat flux densities. Configurational flexibility enables two approaches to radiative power exhaust to be examined in W7-AS: (1) A boundary photosphere of the core plasma may be established by injecting impurities from the edge into limiter bounded magnetic configurations. Experimentally, a steady state radiation level is adjusted by nitrogen injection from a valve, feedback controlled via VUV line emission. (2) Impurities may specifically be injected and eventually confined into magnetic boundary islands in order to provide a local heat sink, topologically separated from the core plasma. In this approach impurities are transiently released from a fast reciprocating erosion probe in a boundary region with magnetic islands. Impurity source and transport properties, the location of impurity radiation and its impact on global confinement and on power balance have been studied for both scenarios. Boundary photosphere studies are presented in detail; first results on the radiating island approach are discussed in ref. [1].

2. Comparative impurity radiation modeling

Figure 1 shows radiation profiles of the low-Z impurities nitrogen and neon and, for comparison, of the medium-Z impurity titanium as calculated with a 1D impurity radiation and transport code (IONEQ [2]). The total radiated power is kept fixed at the arbitrary value $P(r_\text{w}) = 100\text{kW}$. Electron density and temperature profiles used in the calculation are those taken at 0.4s from the nitrogen fuelled discharge in Fig.3, which is simultaneously heated by 300kW ECRH and 650kW NBI (line averaged density $n_0 = 8\times10^{19}\text{m}^{-3}$, $T_e(0) = 1000\text{eV}$, $T_e(a) = 25\text{eV}$, major radius $R = 2\text{m}$, effective minor radius of the last closed magnetic surface (LCMS) $a = 0.18\text{m}$ and of the wall $r_\text{w} = 0.20\text{m}$, impurity diffusion coefficient $D = 0.5\text{m}^2/\text{s}$).

For nitrogen and neon the maximum of the emissivity $Q(r)$ is localized close to but inside the LCMS. A second maximum deeper inside the confinement region significantly contributes to the power loss $P(r)$, which explains the experimentally observed degradation of energy confinement even in the case of nitrogen (see below). For the same total power loss the central concentration is about twice as high for nitrogen, but the central dilution as expressed by the increase of $Z_{\text{eff}}$ is the same as for neon. In view of the requirement of low radiation within the plasma core, nitrogen, radiating closer to the edge than neon and being completely ionized in the center, should be favourable for relatively small devices such as W7-AS.

The titanium radiation is more smoothly distributed over the plasma cross section. A strong reduction of the effective heating power $P_{\text{heat}} - P_{\text{rad}}$ has to be expected in the core. Therefore, medium Z impurities are not suited for edge cooling in W7-AS under the given discharge
conditions, i.e. limiter bounded plasmas with modest electron temperatures, unless penetration into the core is prevented, e.g. by trapping of the impurity ions in edge localized islands or by screening due to a high edge density and/or diffusivity.

3. Nitrogen recycling and fuelling

Radiation feedback control requires effective impurity particle exhaust. Because impurities cannot be effectively removed by external pumps during a plasma discharge in W7-AS at present, particle removal must rely on wall pumping, i.e. on a recycling coefficient significantly below one. Therefore nitrogen, which is expected to have a low recycling coefficient, was finally chosen as impurity rather than neon, which has previously been used in tokamaks equipped with a pumped limiter [3] or a divertor [4].

Nitrogen is injected into the plasma from an absolutely calibrated valve at the outboard side of the wall. The gas flow can be adjusted either to a programmed flow rate or, by feedback control, to produce a programmed radiation level. The intensity of the 765Å line of NIV, measured along a central chord by a VUV-spectrometer, has been used as the feedback parameter. The gas flow is monitored by the 4631Å line of NI, the spectrometer viewing along the gas flow direction.

Wall pumping of nitrogen proved to be sufficiently high for feedback control. Decay times after turning off the gas feed at the end of a phase of constant flow rate were of the order \( \tau_1^* = 25 \text{ms} \) for Be-like NIV (ECRH heated discharge, \( P_{\text{ECRH}} = 260 \text{kW}, \langle n_0 \rangle = 4 \times 10^{19} \text{m}^{-3} \)). The central chord soft-X radiation exhibited a bi-exponential decay. The fast decay time of about \( \tau_2^* = 25 \text{ms} \) is attributed to H- and He-like nitrogen. The slow decay time of about \( \tau_3^* = 70 \text{ms} \) is attributed to the transport of fully ionized particles from the plasma center, becoming radiative after recombination into H- and He-like ions.

In steady state, the probability \( f_z \) that an injected impurity atom is confined in the plasma in the ionization state \( z \) is \( f_z = N_z/(\tau_z^* \Phi) \), with the impurity flow rate \( \Phi \), the effective ion confinement time \( \tau_z^* \), and the number of ions \( N_z \). We use \( \tau_j^* = \tau_1^* \) for ions up to Be-like, \( \tau_2^* \) for H- and He-like and \( \tau_3^* \) for fully stripped ions. For the previous discharge parameters an increase in soft-x radiation by an amount of 25kW is observed when nitrogen is puffed at a rate of \( \Phi = 2.3 \times 10^{20} \text{atoms/s} \). This increase can be modeled (\( D = 0.2 \text{m}^2/\text{s} \)) by a total number of \( 1.4 \times 10^{18} \) nitrogen ions, where stripped ions contribute 57%, H- and He-like ions 36%, and ions up to Be-like 7%. The central nitrogen concentration is 2.5%. From these data a total fuelling efficiency \( \Sigma f_z \) of only 15% is calculated.
Because the fuelling efficiency is low, a large amount of nitrogen has to be introduced into the vessel, which due to the high sticking probability leads to the shot-to-shot build-up of an intrinsic nitrogen reservoir. However, this reservoir settled at a stationary radiation level an order of magnitude lower than that resulting from injection. Short (10 min) helium glow discharge cleaning was not effective in reducing this internal reservoir, but it decreased after stellarator discharges without external nitrogen puff. Therefore long-term contamination of the vessel with nitrogen was not observed.

4. Experimental radiation profiles
Radiation profiles across poloidal cross sections are measured by two 30-channel bolometer arrays, two 36-channel soft-x ray cameras and, spectrally resolved, by a VUV-spectrometer combined with a rotating mirror. On the inboard side the NIV and NV VUV-radiation shells are close to the LCMS in accordance with the expectations from modeling. On the outboard side these shells extend much further outward. This poloidal asymmetry can be related to the local perturbation introduced by the gas puff, which is located at the outboard wall in a toroidal distance of only 8° from the VUV observation plane. The evident difference in Fig. 3 between the time evolutions of the NIV-intensity, which is used for feedback control, and the radiation power is probably caused by the same local perturbation.

The poloidal distribution of the total radiation is shown in Fig.2. In the equatorial plane (z=0) the LCMS is located at R = 178 cm (inboard) and R = 215 cm (outboard). Nitrogen fuelling strongly increases the edge radiation. In a reference discharge without nitrogen injection, intrinsic low-Z impurities (boron and carbon, W7-AS is operated with graphite limiters and boronized wall) provide an already hollow radiation pattern, but at much lower level.

5. Power balance
The radiated power $P_{\text{rad}}$ and the power $P_{\text{lim}}$ conducted to the limiters are derived from the local bolometry and thermography [5] measurements assuming toroidal symmetry. Time traces for the discharge with nitrogen puff and the unperturbed reference discharge are shown in Fig. 3. In the quasi steady phase ($\frac{dW}{dt} \ll P_{\text{heat}} = 950$ kW) $P_{\text{rad}}$ and $P_{\text{lim}}$ are strictly correlated such that the sum is constant. The power accountability exceeds 85%, which gives confidence in the symmetry assumption. The radiated power fraction can be increased from 20% to more than 60% causing a corresponding reduction of the limiter load. The electron temperature near the LCMS as measured by a Langmuir probe decreased from 75 to 25 eV, the edge density increased by up to 30%. During the injection phase, both, the plasma stored energy and the central electron temperature degrade continuously with increasing radiation. The discharge recovers instantaneously on the nitrogen confinement time scale when the nitrogen puff is stopped. If the radiation level is increased further the discharge becomes radiatively unstable and starts to collapse softly from the outside. The maximum fraction of radiated power which
Fig. 3: Diamagnetic energy, central electron temperature, NIV intensity, radiated power and power conducted to the limiters for a nitrogen fuelled discharge and an unperturbed reference discharge. Heating power is 950kW.

could be maintained in quasi steady state depended on heating power and increased from 33% at 180kW (ECRH) to about 60% at 1MW (ECRH+NBI).

The nitrogen radiation shell occupies a significant fraction of the confinement volume and reduces the effective heating power $P = P_{\text{heat}} - P_{\text{rad}}$. Therefore the degradation of the stored plasma energy can qualitatively be explained by the global W7-AS energy confinement scaling [6] $\tau_E \sim P^{-0.55}$, i.e. $W \sim P^{0.45}$.

Electron heat transport in the core seems not to be affected. The heat diffusivity $\chi_e(r)$ from a local transport analysis is similar for both discharges, with and without nitrogen puff. The analysis accounts for a radially distributed heat sink by using the radiation profile of Fig.1 scaled to the total radiation power from the experiment. The temperature profile predicted for the nitrogen fuelled discharge with $\chi_e(r)$ from the reference discharge and additional radiative losses agrees well with the experimental one (Fig. 4). For the prediction the edge temperature is set to the experimental value.

5. Summary and conclusion

Nitrogen has been proven to be appropriate for radiative edge cooling studies in relatively small devices such as W7-AS which are not equipped with active pumping capabilities. Recycling and radiation properties are in accordance with expectations. Long-term contamination with nitrogen was not observed in spite of the high sticking probability. Injection into limiter bounded plasmas provided hollow radiation profiles, reduced the limiter load and strongly cooled the edge plasma. Electron transport analysis does not indicate enhanced heat transport. The degradation of the stored plasma energy with increasing radiation power is rather explained by a reduction of the effective heating power. This seems to limit the radiative stability and thus the maximum radiated power fraction which could be achieved in steady state (up to 60%, increasing with heating power). Detachment from the limiters ($P_{\text{lim}} \rightarrow 0$) could not yet be realized in steady state.

References
Frequency shift of reflectometry signals
due to rotation of density turbulence in W7-AS

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Introduction

Reflectometry provides information about density fluctuations from a measurement of the phase changes of a mm-wave reflected at the cut-off layer in the plasma. Under turbulent plasma conditions an unidirectional drift of the measured phase is observed which cannot be explained by a realistic movement of the cut-off layer due to density changes. An understanding of this phase drift is essential for the interpretation of phase and amplitude of the reflected wave in terms of density turbulence. To clarify the situation two-dimensional simulations of the interaction between the mm-wave and the turbulent plasma have been performed. They show that if parts of the wavefront reflected from different regions in the plasma interfere destructively at the receiving antenna sudden phase changes appear in combination with a drop in the signal amplitude. A preferential direction of this changes - i.e. a phase drift - results, if the density structures are rotating with the plasma. In addition some asymmetry between the reflecting structures and the wavefield must exist. The validity of such a model based on plasma rotation has been investigated experimentally with the reflectometer at W7-AS.

Experiments

The reflectometer system at W7-AS combines heterodyne detection, a high dynamic range and bandwidth with operation at small wavelengths (X-mode propagation). The two poloidally separated antennas (θ=+6° and -6°) for launching and receiving the signal use Gaussian beams focused to a beam waist of 2cm in the plasma. The symmetry axis between the antennas is in the equatorial plane oriented towards the torus center at a toroidal position with elliptically elongated flux surfaces. The cut-off surfaces are almost perpendicular (87.4°) to the probing direction. An Amplitude Modulation system integrated into the reflectometer provides a time delay measurement which is used to obtain density profile information. All signals, carrier phase, time delay and signal amplitude, are sampled at 10 MHz. These characteristics make the system suitable for a detailed analysis of the reflectometer response to plasma density perturbations. From a previous study instrumental reasons, e.g. insufficient sampling or an intermittent loss of signal power, can be excluded as the origin of the observed drift.

The observed phase drift strongly depends on plasma conditions. The most pronounced effect is its sudden disappearance in the H-mode. Under typical L-mode conditions the magnitude of the phase drift shows a characteristic radial profile: it is
positive for positions outside the separatrix and negative within the confinement region with a maximal red shift of a few 100 kHz. This red shift decreases towards the plasma center. If the drift is strong the phase spectrum has a 1/f dependence and the measured amplitude and phase signals show low coherence. In contrast, if the unidirectional phase drift is negligible in comparison with the symmetric phase fluctuations, the spectra differ from a 1/f dependence. In this case for frequencies f > 50 kHz amplitude and phase signals are coherent with a phase shift between them very close to π/2.

In order to test the influence of the antenna geometry on the observed drift transmitting and receiving antenna have been interchanged on a shot to shot basis using a variety of different plasma conditions (toroidal magnetic field of 1.25 / ±2.5 T, heating with 400/800 kW ECRH and 0.5 MW NBI, average density between 2 and 14 10^{19} m^{-3}, rotational transform 0.34 and 0.53): The observed phase drift does not change significantly when transmitting and receiving antenna are interchanged.

The correlation between the phase drift and poloidal plasma rotation has been investigated for two types of ECRH heated discharges which display different rotation profiles (Fig. 1): (1a) ECRH heating 400 kW off axis + 400 kW on axis and (1b) ECRH 400 kW off axis only. The poloidal rotation profile of the plasma is obtained as a sum of ExB and pressure driven rotation. The radial electric field is measured from the Doppler shift of impurity emission lines (B IV). As can be seen in Fig.1, for the two plasma conditions the radial profile of the phase drift qualitatively corresponds with the poloidal plasma rotation profile. For pure off-axis heating (Fig. 1b) the electron temperature and its gradient have very low values inside the heating location (r_{eff}~10 cm). Therefore the diamagnetic contribution to plasma rotation is very low at these positions, a feature that is also observed in the values of the phase drift.

An inversion of plasma rotation has been achieved by inverting the magnetic fields of the stellarator. All other plasma parameters are kept constant: Density and temperature profiles measured with Thomson scattering, Li-beam, reflectometry and ECE respectively show no variation as the magnetic field is reversed. Impurity rotation (B IV) changes sign confirming the inversion of the ExB velocity. An example of the observed phase drift is given in Fig. 2: For all probed radial positions the drift is inverted as the plasma rotation is reversed. In some discharges the absolute value of the drift measured for positive and negative magnetic fields nevertheless differs by up to a factor of 2.

A detailed analysis of the phase changes shows that the inversion of the drift is related to the shape of the measured phase fluctuations. Figs. 3 gives a detail of the phase and amplitude of a reflected signal for two corresponding discharges with positive and negative magnetic field. For example with positive B (Fig. 3b) during a fluctuation the measured phase increases faster than it decreases and the minima in the amplitude occur during the phase increase. This observation that the amplitude minima correlate with the bigger rates of phase change is consistent with the simulations. A coherency analysis shows that the coherence
between phase and amplitude is high for frequencies 50 kHz < f < 2 MHz and the phase shift between both signals is ±π/2 for positive and negative magnetic field respectively.

As during this period the phase does not always recover to its previous value, in the long term a drift in negative direction is obtained. This intermittent behaviour can be explained taking into account that the phase responds non linearly to changes in the local density gradient induced by the fluctuations. This local gradient is a function of the average one, the amplitude and the wavelength of the fluctuations. An intermittent phase drift appears if for certain time intervals, the turbulent structures are such that the microwaves reflected at them interfere destructively at the antenna. During the rest of the time the amplitudes of the density turbulence are lower or their wavelengths are larger therefore all power received at the antenna is reflected at a smoother surface i.e. at the same distance to the horn. Another explanation would be that the intermittence occurs due to small turbulence structures, that appear and disappear.

Discussion

The experimental results obtained with the reflectometer at W7-AS are compatible with an explanation of the observed phase drift based on density structures rotating with the plasma. As the direction of the drift can be inverted by inverting plasma rotation radially propagating density bursts can be excluded to be the main origin of the phase drift.

For the origin of the asymmetry in the reflected wavefield necessary to explain a net phase drift some experimental clues can be given: A vertical displacement of the plasma larger than 2-3 mm with respect to the antenna axis can be excluded from the lack of profile change as the magnetic field is reversed. Following the 2D WKB code calculations this small displacement can not produce the required asymmetry unless the turbulent structures have very high amplitude or very short poloidal wavelengths. The same holds for the small 2.6° misalignment of the horn axis from the normal to the cut-off surfaces. According the 2D simulations the poloidal separation of emitting and receiving horns could introduce the required asymmetry. Nevertheless the experiments show that the phase drift does not change significantly as the horns are interchanged. Therefore the experimental results would be consistent with an explanation of the phase drift based mainly on an asymmetry in the turbulent structures themselves. As the observed phase drift direction is inverted when inverting the magnetic fields in this case the asymmetry must be independent on the rotation direction. In order to obtain more conclusive results 2D full-wave simulations are currently under way.

References

**Fig. 1:** Radial profiles of poloidal plasma rotation (ExB-component obtained from spectroscopy and diamagnetic drift) and of the phase drift for two types, a), b) of ECRH heated discharges.

**Fig. 2:** Frequency shift of the reflected signal as a function of cutoff layer position for discharges with normal and inverted magnetic fields. Plasma rotation changes as B is inverted while all other plasma parameters are kept constant.

**Fig. 3:** Time traces of reflectometer phase and amplitude for positive and negative B-fields. The asymmetry of phase and amplitude signals changes sign as the magnetic field is reversed.
Ion Cyclotron Resonance Heating Experiments on the Stellarator W7-AS

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Ion cyclotron resonance heating (ICRH) has, for the first time, successfully been demonstrated on the stellarator W7-AS. A novel broad antenna [1,2] designed to excite a narrow spectrum of fast waves was used. Two different heating scenarios were investigated: second harmonic heating of neutral beam heated hydrogen plasmas and hydrogen-minority heating of ECRH deuterium plasmas. Both scenarios showed plasma heating without a significant concurrent increase in plasma density or impurity radiation loss. In addition, it was possible to sustain the plasma with ICRH alone.

The ICRH antenna, shown in Fig. 1, is located on the high field side in the elliptical cross-section of the non-axisymmetric plasma. It has four feeders that allow operation in 0- and π- phasing. Typically it is operated in π-phasing. Then the poloidal current has an almost sinusoidal distribution in the toroidal direction and excites a narrow \( k_{\parallel} \approx 6 m^{-1} \) spectrum of fast waves. During the Spring 1995 opening of the torus vessel the feeders to the antenna, shown in Fig. 2, were closed off against plasma penetration. This eliminated anomalously high loading of the antenna during plasma operation observed previously [3] and increased the maximum rf-voltage at which electrical breakdown (arching) occurred. Most plasma targets had \( \iota \approx 0.33 \) and were shaped by inside limiters. In those plasmas the distance from the antenna to the fast-wave cutoff is about 6 cm and the observed antenna plasma loading is only about 0.5 \( \Omega \). Voltages of up to 55 kV for 400 msec, corresponding to a maximum power of about 400 kW at the antenna, have been achieved after extensive conditioning. Real time visual observation of the antenna during plasma operation and inspection after the vacuum break confirmed that arcs did not occur in the antenna, but only in the feeders and in the transmission lines.

In the second harmonic hydrogen heating scenario with a neutral beam heated target plasma an increase in the diamagnetic energy of about 10 % (0.6 kJ) was obtained. The central hydrogen temperature was about 800 eV and central electron densities were \( 6 \times 10^{19} m^{-3} \). One estimates
that about 60% of the power $P$ radiated from the antenna is found in the plasma, if $P$ is the
generator power reduced by the ohmic losses in the antenna and $P^{-0.6}$-scaling of the energy
confinement time is invoked. Under good wall conditions, the plasma density could be kept
constant during the rf-pulse, even though the $H_{\alpha}$-observation indicated enhanced outgassing
at the antenna. The impurity radiation inferred from the bolometer did not increase. An
increase in the flux of hydrogen atoms with energies between 10 keV and 33 keV was observed;
however, no significant increase in the bulk hydrogen temperature was observed. Maximum
heating occurred if the location of the second harmonic resonance coincided with the center of
the plasma; almost no heating occurred if the resonance was outside of the plasma. The antenna
loading was independent of the location of the resonance even though a rf-probe (located half-
way around the torus) detected a wave signal only if the resonance was outside of the plasma.
Heating at 0-phasing showed similar increases in the diamagnetic energy as in π-phasing and
no enhanced impurity radiation. No significant heating was observed in ECRH plasmas of the
same density, presumably because the hydrogen temperature of the target plasma was too low
(about 350 eV).

In the H-minority heating scenario with an ECRH target deuterium plasma an increase in the
diamagnetic energy of about 15% (1 kJ) was obtained. This corresponds to absorption of
about all of the radiated power $P$. Fig. 3 shows a typical example of a plasma shot. The
spectroscopically estimated $H/D$ ratio was about 10%. The line-of-sight averaged deuterium
temperature rose from 300 eV to 400 eV; the central electron temperature rose slightly. Energetic
hydrogen atoms with energies up to 33 keV were observed. The impurity radiation did not
increase.

In this heating scenario it was possible to sustain an ECRH-created plasma with ICRH alone
for as long as 500 msec. The duration of the ICRH-only phase of the plasma was solely limited
by arcing in the transmission lines. An almost steady state condition could be obtained about
200msec into the ICRH-only phase of the discharge. Typical parameters were diamagnetic
energy of 2 kJ, average electron density of $4 \times 10^{19} \text{m}^{-3}$, central electron temperature of 300 eV,
and central deuterium temperature of 350 eV.

An example of an ICRH sustained plasma is shown in Fig. 4 starting at 400 msec. The generator
frequency and the approximate $H/D$-ratio were such that both, the H-resonance and the ion-ion-
resonance were located inside of the plasma volume, as shown in Fig. 2. The average electron
density first rose by increased outgassing of the antenna but returned towards the initial value
near the end of the ICRH plasma. The central electron temperature dropped rapidly within
an energy confinement time and then stayed constant throughout the ICRH phase. The total radiation measured with bolometers stayed constant, even though an accumulation of iron and chromium could be inferred from VUV observation; yet soft X-ray measurements indicated that $Z_{\text{eff}}$ did not increase. We can therefore conclude that, at least in the range of parameters investigated, the ICRH sustained plasma is not hampered either by uncontrolled density increase or by enhanced impurity production.

The plasma density profile, measured with Lithium beam diagnostic, Langmuir probes, microwave reflectometry and Thomson scattering, was narrower and had steeper edges than in comparable ECRH heated plasmas. The transition between these two profiles occurred within approximately one energy confinement time. Further narrowing of the plasma was observed on a longer time-scale. The resulting increase of the distance from the antenna to the fast wave cutoff could explain the decrease of the antenna plasma loading and therefore the decrease in diamagnetic energy. A radial electric field of about -1.5 kV/m built up at the beginning of the ICRH phase of the discharge, presumably due to increased high-energy hydrogen losses as indicated by CX measurements.

References:
Figure 1. ICRH antenna shown without Faraday screen.

Figure 2. Poloidal cross-section through the ICRH antenna. Resonances and cutoffs shown for the ICRH plasma in Fig. 3.

Figure 3. Time trace of shot 33882. Ion temperatures inferred from unweighted line-of-sight average of CX fluxes. $B_T = 2.5T, \iota = 0.34$.

Figure 4. Time trace of shot 33634. Ion temperatures inferred from unweighted line-of-sight average of CX fluxes. $B_T = 2.5T, \iota = 0.34$. 
1. Introduction
The Li beam diagnostic on W7-AS /1/ allows the measurement of neutral density $n_0$ outside the plasma column as well as electron density profiles of the main plasma by observation of the collisionally induced Li II line radiation (671nm) from an injected high energy neutral Li-beam (20-66keV /0.5-3.0mA). Central profiles for higher densities are achieved by incorporating line integrated data from the HCN interferometer /2/. Thus highly accurate density profiles from the plasma middle to the walls are available. Using $n_e$-profiles and $n_0$ derived from the Li-beam in combination with $H_\alpha$ intensities measured at several points around the vessel, the Monte-Carlo code EIRENE is implemented to determine the important neutral sources as well as the absolute neutral density distribution.

2. Density profile reconstruction
For peak densities $n_{00} < 4 \times 10^{19} \text{ m}^{-3}$ profiles spanning the entire outer radius can be reconstructed by the Li beam diagnostic. Central profiles for higher densities are achieved, under the assumption of flat profiles, by incorporating data from HCN interferometry.

The HCN interferometer on W7-AS measures line densities along three parallel lines of sight in the $\varphi=29^\circ$ plane (approximate elliptical plasma cross section, cf. fig. 2.1). At low densities these data can be used to cross check the Li beam data. To this end the reconstructed Li density profiles are mapped along the magnetic surfaces using the TRANS-database to the plane of the HCN interferometer, integrated along the line of sights, and directly compared. Using the electron temperature profile as measured by Thomson scattering in the Li beam density reconstruction algorithm /3/ and a central beta consistent with the derived profile for the field line mapping, excellent agreement is found, cf. fig. 2.2. For higher central densities $n_0 \leq 1 \times 10^{20} \text{ m}^{-3}$ the entire gradient region is still accessible by the Li beam diagnostic. The effective plateau value of the density profile can be calculated by subtracting the contribution of the plasma edge from the central HCN channel. A cubic spline approximation is used to construct a smooth transition between the edge density profile and the derived plateau value. To take into account possible uncertainties in the Li beam density profile, the plateau value is varied in such a way that the line integral of the smoothed density profile fits the line density of the HCN interferometer to within a tolerance of less than 2%. The outermost channel of the HCN interferometry is used to check the field line mapping. For the EIRENE calculation below a low iota ($I_\alpha=0.34$, no islands at the plasma edge), medium density (small beta effects) shot (#33359) is chosen. The derived density profile for this shot together with the edge density profile measured by the Li beam diagnostic is shown in fig. 2.3. Measurements of this quality can be routinely carried out on a ms time scale.

3. Neutral density measurements
The neutral density is registered in the injection port by observing the Li$_2$p line radiation from the Li beam induced by collisions between Li beam atoms and neutral particles. The observation volume is at a point 0.55 m from the plasma surface, thereby minimizing any interference from background light at the Li2(p) wavelength while maintaining a good vacuum conductance to the main vessel. The signal is calibrated by injection of the Li beam into the discharge chamber at known molecular gas pressure and applied magnetic field. Since EIRENE calculations indicate that moleculles make up 99% of the neutral density within the injection port, this calibration may be directly applied to measurements performed during plasma discharges. For the discharge under investigation, see the EIRENE calculation below, the neutral density is determined to be $n_0 = 3.6 \times 10^{17} \text{ m}^{-3}$ (1.5-10^{-5} mbar).
4. H_α measurements

H_α intensities are recorded at different toroidal positions (cf. fig. 5.5): (1) spatially resolved by a diode array at the inner limiter, (2) at φ=10°, (3) at φ=18°, the position of the Li beam diagnostic and (4) at φ=23°. Data for positions 2-4 are absolutely calibrated via an Ulrich sphere (e.g. \( H_\alpha(\phi=18^\circ) = 1.8 \times 10^{18} \text{ photons m}^{-2}\text{s}^{-1}\text{sr}^{-1} \)) and used in the EIRENE calculations.

5. EIRENE simulation of H_α and n_0 measurements

The neutral gas distribution plays a crucial role for source terms entering into the particle balance of the plasma edge and hence is an important component of edge investigations. To determine the absolute neutral density in the module of the Li beam, the neutral density distribution is calculated with the 3D Monte-Carlo code EIRENE /4/, using \( n_e(r) \) profiles and \( n_0 \) from the Li beam as well as the H_α intensity measured at 4 positions including the Li beam port and the poloidal H_α distribution at an inboard limiter. With these independently measured quantities \( n_0 \) and H_α a consistency check of the EIRENE modelling of the neutral source locations is also possible. To properly accommodate the \( n_0 \) measurement located deep inside a port, the Li beam port is also simulated as shown on the computational grid in fig. 5.1. Periodicity is assumed in the toroidal direction, i.e. each of the 5 modules is treated identically.

For clear conditions concerning neutral sources a discharge without islands (\( i_g=0.34 \)) is considered, where the plasma has been shifted towards the inboard limiters (\( B_z = 18\text{mT} \)) so that they comprise the main neutral source. The neutral source distribution is taken to be the same as the H_α intensity profile measured along the plasma-facing side of an inboard limiter.

This assumption yields self-consistent results: in fig. 5.3 the comparison between experimental and calculated H_α profiles shows good agreement over the entire limiter. Fig. 5.4 illustrates the corresponding lower inboard limiter and H_α line of sights, together with the neutral atom distribution. Since the distance of the Li beam to the inboard limiter is about 0.5 m toroidally and 150° poloidally, it is not clear a priori, which source dominates the local neutral density, i.e. the limiters or the global plasma-wall interaction. The scaling of the neutral particle distribution from the limiter source to its absolute H_α intensity shows that at the Li beam plane this source strength is too low for the corresponding H_α signal. This becomes clear from the toroidal variation of the neutral density shown in fig. 5.5: Between the limiter and Li beam plane the atomic density decays by 2 decades. Therefore a second source at the wall is introduced, which for simplicity is assumed to be a point source in the Li beam plane. It turns out that one has to put its location at the upper part of the so-called helical edge (corresponding to the upper tip of the plasma in fig. 5.1), which is expected to be the principal plasma-wall interaction area away from limiters. By scaling the point source neutral distribution with the H_α signal at the Li beam, \( n_0 \) in the Li beam port and the next further H_α are predicted by the calculation to within 50%, whereas with other source locations the differences are more than an order of magnitude, if a reasonable solution exists at all. This shows that the helical edge defines the dominant region for the global plasma-wall interaction. In fig. 5.2 the toroidally and poloidally averaged atomic densities of the two sources are compared: inside the limiter the limiter source dominates the neutral density by a factor of 10, but outside the wall source is even higher than the latter.

6. Conclusions

Employing the Li beam \( n_e(r) \) profiles and H_α intensities it has been possible to find a self-consistent 3D neutral source distribution in EIRENE showing that on the limiter the source profile corresponds roughly to the H_α intensity profile and that the helical edge is mainly responsible for the global plasma-wall interaction. In addition the neutral density in the Li beam port could be modelled for the first time, giving agreement within 50% to the experimental \( n_0 \) value from the Li beam.
Fig. 2.1 Magnetic flux surfaces for discharge #34540 with line of sights for the HCN interferometer.

Fig. 2.2 Line densities of the HCN channels indicated in fig. 2.1 (solid lines) compared to the line integrals calculated from the Li beam data (symbols) for discharge #34540.

Fig. 2.3 Reconstructed profile using Li-beam edge density profile (symbols) and the central HCN chord for discharge #33359.

Fig. 5.1 Computational grid and diagnostics at the Li beam phi plane. The Li-beam $n_e$ profile along the beam line is directly mapped onto the grid flux surfaces. Toroidally the grid has a resolution of 50 cells per module.

Fig. 5.2 Globally averaged atom density profile of the inboard limiter and the helical edge wall source with the Li beam $n_e$ and Thomson temperature profile. The shadow line indicates the limiter position.

Fig. 5.3 Experimental and calculated $H_\alpha$ profiles at the lower inboard limiter. The tile index varies from bottom to top.

Fig. 5.4 Lower inboard limiter and $H_\alpha$ profile from the array as main neutral source.

Fig. 5.5 Toroidal atom density distribution of the limiter source on the flux surface in front of the inboard limiter for the half of one module. Higher values correspond to brighter colours. The upper picture shows along the poloidally averaged atomic density the toroidal positions of the inboard limiter (1), the Li beam plane (3) and two other $H_\alpha$ locations (2,4).
Stability of W7-AS Configurations with Reduced Vacuum Magnetic Well

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Introduction: Wendelstein 7-AS is a modular low shear stellarator with 5 field periods, a major radius of about 2 m and a minor radius of about 18 cm. W7-AS is an intermediate step towards a fully optimized low shear stellarator like the projected W7-X. The Pfirsch-Schlüter-currents in W7-AS are already reduced by the smaller average toroidal curvature compared to a classical stellarator, although not in the extend as in W7-X. This partial optimization results in better equilibrium properties (reduced Shafranov shift), and in a better stability behaviour since the PS-currents are driving terms for ideal (Mercier-) and resistive interchange modes. Stability at low $\beta$ is usually provided by a vacuum magnetic well of up to 2%. This is in contrast to the torusatron/heliotron line of stellarators which are stabilized against Mercier modes by a strong shear inevitably leading to a large number of rational surfaces inside the plasma. A low shear stellarator tries to avoid low order rational surfaces. With respect to resistive interchange modes torsatrons/heliotrons are unstable due to a magnetic hill in their magnetic configurations at least at outer radii. Resistive interchange modes are thus thought to cause most of the turbulent transport in the gradient regions of such machines [1].

To study the importance of interchange stability for W7-AS an investigation of the predicted ideal and resistive interchange stability limits had been started for the accessible magnetic configurations of W7-AS [2]. We extend this study in particular to configurations with vanishing vacuum magnetic well. These are more unstable with respect to Mercier and resistive interchange modes than the configurations for low and medium $\beta$-experiments. We discuss this stability analysis in comparison with experimental results.

Configurational space of W7-AS: A magnetic configuration of W7-AS is determined by the currents in the four coil systems, namely in the modular coils ($I_m$), in the large special coils ($I_s$) located in the region of strongest toroidal curvature ("elliptical" cross section), the toroidal field coils ($I_t$) and in the vertical field coils ($I_v$). Therefore, the magnetic configuration is defined by three coil current ratios: $I_t/I_m$ (the vertical field) is related to the position of the magnetic axis and reduces the vacuum magnetic well in inward shifted configurations. $I_s/I_m$ (additional toroidal field) can be used to control the rotational transform $\ell$, and $I_v/I_m$ determines the toroidal ripple, $r$, of the magnetic field strength as defined by its Fourier components in Boozer coordinates. The "standard" configuration is defined by $I_s/I_m = 1$ and has $(r \approx 0)$. Configurations with $r < 0$ have an increased magnetic field strength in the "elliptical" plane. Thus, trapped particles are shifted out of the region of strong curvature, so that trapped particle modes are expected to be not important. Additionally, a negative toroidal ripple reduces the vacuum magnetic well in W7-AS.

Equilibrium calculations for stability analysis: We analysed inward shifted "standard" configurations at $\epsilon_{vac} \approx 0.34 - 0.35$, and about 0.4, and "ripple" configurations with $r = -20\%$ at $\epsilon_{vac} \approx 0.34 - 0.35$ where high-$\beta$ discharges had been performed. In earlier experiments top and bottom limiters in the "elliptical" plane determined the plasma boundary, and vertical fields with $B_z/B_0 \leq 0.011$ had been applied for position control ($B_0 =$ mean magnetic field strength). Experiments in the last campaign without these limiters and with the upgraded NBI (8 injectors instead of 4, i.e. almost doubled input power) required higher vertical fields. Values of $B_z/B_0 \approx 0.012 - 0.025$ were used in both the "standard" and the "ripple" configuration. The plasma boundary was determined by new limiters on the inner torus wall.

Free-boundary equilibria were calculated with the NEMEC-code [3] assuming pressure profiles
proportional to $(1 - s)$ and $(1 - s)^2$, $s$ being the normalized toroidal flux ($s \approx (r_{\text{eff}}/a_{\text{ref}})^2$), resulting in $\beta_0 / \beta \approx 2$ and 3, respectively ($\langle \ldots \rangle$ denotes the volume average). The first profile shape is more typical for heating with NBI, the second one for ECR heating. The plasma boundary was chosen to touch the inner limiters for the converged equilibrium. In the calculation we neglected toroidal net current densities, i.e. the bootstrap current, ohmically and NBI driven currents. This is reasonable for high density, low temperature plasmas at high $\beta$ with balanced NBI. Fig. 1 shows that the PS-currents have considerable influence on the plasma position and the $\iota$-profile at high $\beta$ due to the low $\iota$-values of W7-AS.

![Diagram](attachment:image.png)

Figure 1: Change of the plasma boundary and axis and of the $\iota$-profile for the standard configuration at $\iota = 0.35$, $B_x/B_0 = 0.025$. The profile shape is (1-s) with $\beta_0 = 0, 0.3, 1, 1.7, 2.4, 3.2, 4\%$.

The vacuum magnetic well of the considered configurations was reduced to marginality by the applied vertical fields. This effect was even pronounced in the "ripple" configuration where a vacuum magnetic hill can be achieved for $B_x/B_0 \geq 0.016$. In the standard configuration vertical fields with $B_x/B_0$ of about 0.025 are required.

Stability analysis of configurations with marginal magnetic vacuum well: For low shear stellarators the stability criterion for resistive interchange modes is in a good approximation given by [4]

$$
(p'V'' - \langle j_1^* \rangle, - \langle j_8^* \rangle, 0) > 0,
$$

if the modes are localized around rational flux surfaces and if toroidal net currents are neglected. Here, $\langle \ldots \rangle = \int d\theta d\phi (\ldots \sqrt{g} / |\nabla s|^2)$ and $(\ldots)' = (\ldots)/ds$. We note that the "peeling modes", which are resonant ideal free boundary modes localized at the plasma boundary have the same stability criterion as the resistive interchange modes at the plasma boundary [5]. Therefore, whenever a resistive interchange unstable region extends to the plasma boundary, "peeling modes" may be present. The less strict Mercier criterion for ideal interchange has an additional stabilizing term depending on the global shear $\iota'$ and is given by

$$
\left[ \iota' + \langle \sqrt{g} B^2 \rangle, \langle \frac{j_8}{B|\nabla s|^2} \rangle \right]^2 + \langle \sqrt{g} B^2 \rangle, \langle \frac{1}{|\nabla s|^2} \rangle, (p'V'' - \langle j_1^* \rangle, - \langle j_8^* \rangle, 0) > 0
$$

with the above assumption of no toroidal net-currents. The idea of shear stabilization in torsatrons and heliotrons is obvious in this formulation.

The stability analysis was performed using the JMC-code [4] which evaluates the Mercier- and the resistive interchange stability criterion on the flux surfaces using Boozer coordinates. The results for the resistive and ideal interchange modes are shown in Fig. 2. Since we are more interested in "general" stability boundaries we did not resolve unstable regions due to low order rationals in the $\iota$-profile [4].

For the "standard" cases at $\iota = 0.35$ and 0.4 we see that for configurations, which are stable at low $\beta$, the resistive interchange unstable region is growing from the boundary towards the inside. For those which have an unstable region even for $\beta = 0$ the magnetic well generated by $\beta$ stabilizes the central part of the plasma leaving the outer fourth to third of the plasma unstable. The configuration with the large toroidal ripple shows qualitatively the same behaviour, but the unstable regions are intruding further towards the plasma center for comparable vertical fields.

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Figure 2: Resistive (upper picture) and ideal (lower picture) interchange boundaries for the different configurations. The arrows point into the stable plasma region.

For the cases, which had been considered previously [2], Mercier stability did not pose any problem. It is different in the cases considered here. For the standard configuration the Mercier criterion starts to play a role at vertical fields of about \( B_z/B_0 \approx 0.016 \) depending on \( \iota \) (stable for \( \iota = 0.35 \), but a small unstable region for \( \iota = 0.4 \)). For low \( \beta \), the unstable region grows from the boundary towards the axis with increasing vertical fields, but is stabilized at higher \( \beta \)-values. Since the finite-\( \beta \) magnetic well is not sufficient to stabilize the resistive interchange modes, the important part for the stabilization of the Mercier modes is the sheared part, which also involves the parallel current density. This last part can be important for Mercier stability in regions where the shear vanishes.

A comparison of the stability with respect to the two different \( \beta \)-profiles shows that the narrower profile is more unstable with respect to resistive interchange modes. This is due to the different finite-\( \beta \) well deepening. However, the stability boundaries with respect to Mercier modes are comparable.

**Experimental results:** High-\( \beta \) experiments had been performed in the configurations analyzed above. Most discharges used vertical fields with \( B_z/B_0 \leq 0.017 \) in the "standard" configuration but a variety of values of the rotational transform \( \iota \) around the above values. In the case of the ripple configurations discharges have been performed only in the ones considered in the stability analysis. Generally, after a start-up with 70GHz ECRH at half field (\( B_0 = 1.25T \)), the NBI was switched on in steps from 2 to 4 to 8 or from 2 to 8 injectors with a total heating power of 2.2MW. The peak densities raised depending on the field strength to values of nearly \( 2 \times 10^{20} m^{-3} \) and electron temperatures of 400eV. These values justify the assumption on the internal toroidal net current densities at least in the moderate to high \( \beta \)-phases of the discharges. The maximum \( < \beta > \)-values were about 1.6 – 1.8% for the standard configurations with peaking factors of about 2.2 regardless of the inward shift.

Even in the configurations with \( r_i = -20\% \) the maximum \( \beta \)-values ranged between 1.4 and 1.6% for 5 and 6 NB-injectors with about the same \( \beta \)-peaking. Experiments using a nonresonant start-up (900MHz Rf-heating) at lower main fields (1.0T and 0.8T) did not lead to the naively expected higher \( < \beta > \)-values. However, as reported in [6], this "\( \beta \)-limit" is mainly due to the degrading heating efficiency of the counter NB-injectors at fields below 1.25T and, additionally,
to the scaling of the maximum achievable density $n_{\text{max}} \sim B_0$.

Although various MHD-activities have been observed in the Mirnov- and soft X-ray diagnostics in the high-< $\beta$ > discharges no general $\beta$-limiting phenomena have been observed. However, in a sequence of discharges in the "standard" configuration with $B/\rho \geq 0.017$ a burst type, broad band MHD-activities occurred, correlated to relaxations in the energy signal and to profile relaxations seen by the SX-ray diagnostic. Because of the reduced resistive interchange stability of these highly inward shifted configurations resistive interchange modes together with the peeling modes could be candidates for this type of activity. The appropriate low order rationals at the plasma boundary, which could be the origin of such relaxation processes, might have been generated by the deformation of the $\rho$-profiles due to the high-$\beta$-values. Nevertheless, the identification is unsure since it is not clear why this type of activity was not observed in the strongly inward shifted ripple cases which are more unstable in this aspect.

Other significant MHD-activities observed are coherent modes at low frequencies ($f \leq 10kHz$) with $k \approx 0$ which are presumably driven by the plasma pressure. Some have been identified by the SX-ray diagnostic with mode numbers (e.g. $(n,m)=$$(1,3),(1,2)$) corresponding to low order rationals of $\rho$ inside the plasma. However, these modes did not show any significant influence on the confinement. Moreover, their appearance does not depend on high < $\beta$ > values, but it is also known to appear at medium values[7]. Coherent mode activity above 15kHz was observed, too, and is thought to be mainly driven by fast beam particles [8].

**Summary:** We have extended the stability studies to magnetic configurations of W7-AS with strongly reduced vacuum magnetic well. Depending on the inward shift (vertical field strength), and/or the toroidal magnetic ripple, these configurations can be Mercier unstable at low $\beta$ stabilizing themselves at higher $\beta$. They are generally also resistive interchange unstable in the outer half of the minor radius. Experimentally, relaxation phenomena have only been observed in the highly inward shifted standard cases. The observed burst type, broad band MHD-activity may be due to resistive interchange or "peeling" modes. However, their identification is far from being sure. Especially, their absence in the more unstable "ripple" configurations is not understood. From these results, the importance of resistive interchange as a soft, $\beta$-"limiting" instability seems to be questionable for W7-AS. However, torsons/trons/heliotrons have a much higher shear at outer radii. This may lead to a stronger radial coupling of resistive modes due to the smaller distances between resistive instabilities on rational surfaces. From the view of the stability analysis, the Mercier unstable regions at low $\beta$ would be more interesting than the high $\beta$ phases. However, since the emphasis in the experiments was on stability limits at high $\beta$, only few data are available to judge on this subject, and a separate experimental investigation of of these cases should be started. Internal toroidal current densities may then be included, and alter the ideal stability boundaries due to the induced shear.

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TEMPERATURE FLUCTUATION MEASUREMENTS WITH ECE ON W7-AS

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Abstract. Electron temperature fluctuations at a relative level of the order of 0.1% in the presence of strong inherent intensity fluctuations of thermal radiation can be measured applying correlation radiometry of electron cyclotron emission. The diagnostic problem and the technique are briefly reviewed. Examples are given which characterize the temperature fluctuations as measured in the core region of purely ECRH heated plasmas with respect to spectrum, relative level, and coherence length. Parameter dependencies are discussed.

1. Introduction

The anomalously high electron heat transport is assumed to be driven at least in part by fluctuations of density, electric field, temperature, and magnetic field. Independent of the driving mechanism, the electrostatic term of the temperature fluctuation driven heat flux is given by \( \dot{q}_e \propto n_e \langle \tilde{E}_\theta \tilde{T}_e \rangle \), where \( n_e \) is the local electron density and \( \tilde{E}_\theta \) the fluctuating poloidal electric field. Temperature fluctuations are measured at the plasma edge with probes. In the plasma core they have been shown to exist applying the correlation methods as described below [Sattler and Hartfuss 1994]. Until now no simultaneous measurement of temperature fluctuations and E-field fluctuations, necessary to calculate the driven heat flux, are possible. Nevertheless a characterization of the temperature fluctuations and the plasma parameter range over which they exist is helpful and necessary as a first step. The comparison of experimental results with numerical simulations of various fluctuations may in future help to gain some insight into their role in electron heat transport.

To measure electron temperature fluctuations the diagnostic set-up must provide a spatial resolution of at least 1 cm both in poloidal and in radial directions, a temporal resolution of up to 1 \( \mu \)s and a sensitivity to relative amplitudes of the order of 0.1%. Radiometry of the electron cyclotron emission can provide the necessary spatial and temporal resolutions but not the required sensitivity to small amplitudes. A detailed discussion of the reasons is given in the literature [Sattler and Hartfuss 1993, Hartfuss et al 1996], it is only briefly repeated here.

The sensitivity is limited by inherent, natural fluctuations of the ECE due to the thermal nature of this radiation. These fluctuations are present even for an emitter of constant temperature. Therefore the sensitivity is not limited by technical problems associated with the detection systems, i.e. the sensitivity of the radiometers in use. The intensity fluctuations of blackbody radiation, which the ECE is assumed to be, are determined by the average value of the detected intensity and the ratio of coherence time of the radiation and integration time of the detection system as determined by the postdetection and the predetection bandwidths of the detector. While in an ECE radiometer the postdetection bandwidth determines its time resolution, the predetection bandwidth determines the radial resolution and the ratio cannot be chosen independently. To reach the necessary space and time resolutions as defined before,
typical values of postdetection and predetection bandwidths are 1 MHz and 500 MHz respectively. The resulting relative noise level set by inherent fluctuations is then about 6% which sets a lower sensitivity limit of an ECE system under quiescent plasma conditions. True electron temperature fluctuations at a level of 0.1% are therefore completely buried in noise.

The limit can be overcome applying correlation radiometry and making use of the spatial coherence properties of thermal radiation and their relation to its intensity fluctuations. If the same ECE emitting plasma volume is viewed by two separated but identical radiometers, the fluctuation level present is the same for the two systems, but if the angle between the two lines of sight exceeds a minimum, the fluctuations in the two radiometers are not correlated. The cross-correlation function of the time dependent radiometer output signals is then zero. If true temperature fluctuations are present in the common emission volume, they are seen by both radiometers resulting in a finite cross-correlation function from which the relative fluctuation level and the fluctuation spectrum can be determined. (A different correlation technique has been developed at TEXT. It makes use of a property of ECE whereby two slightly separated frequency channels correspond to indistinguishable radial positions of the temperature profile [Cima et al. 1993].)

A correlation radiometer of this kind has no time resolution because a large amount of data is necessary to determine the cross-correlation function with sufficient precision (< 0.1%). Typically a few $10^6$ data points are necessary which demands at a sampling rate of 1 MHz for stationary plasma conditions for longer than 1s. If these conditions cannot be sustained, several equal shots must be combined. The k-resolution in radial direction is determined by the radial resolution of the ECE radiometers (set by the predetection bandwidth) and is about 2.5 cm$^{-1}$ for this experiment. In poloidal direction it is determined by the spot size of the ECE viewing optics which is formed by the slim beam waists of Gaussian optics as provided in both sightlines by means of enlarged elliptical mirrors. This resolution is enhanced over the former W7-AS intensity interferometer [Sattler and Hartfuss 1994]. The resulting resolution in poloidal direction is now 2-3 cm$^{-1}$. This means that for both components radial as well as poloidal only temperature fluctuations with k-values below 2-3 cm$^{-1}$ contribute to the measured temperature fluctuation level.

2. Experimental Conditions and Results

In each of the two crossed sightlines 6 radial channels are installed allowing for correlation radiometry at 6 radial positions covering the region of about 5 to 10 cm in effective radius corresponding to $0.3 \leq r/a \leq 0.6$ on the high field side of the normalized plasma minor radius. For on-axis ECRH this corresponds to the gradient region of the electron temperature profile with typical electron temperatures of the order of 1 keV in the experiments described below. The density profiles at W7-AS are broad under these conditions which provides almost constant electron density conditions in the observation region. Figure 1 gives typical profiles and sketches the radial positions accessible. The results discussed below are obtained for a B-field on axis of 2.5 T with both 70 GHz and 140 GHz ECRH under 1st harmonic o-mode (O1) and 2nd harmonic x-mode (X2) conditions respectively. The ECE correlation radiometers detect the 2nd harmonic x-mode ECE. The frequency range covered lies between 143 and 148 GHz.
2.1 Fluctuation Spectrum

Figure 2 gives the crosspower spectra for the 4 outermost channels. The spectra shown are typical for all spectra obtained under on-axis ECRH heating conditions. Qualitatively they are independent on heating power and electron density. The spectra are characterized by large fluctuation power below about 20 kHz and a smaller but broader component extending into the 100 kHz range with a broad feature around 50 kHz. It will be shown below that the two components behave physically different, as already observed in earlier experiments [Hartfuss et al 1993]. But due to the improved poloidal resolution of the detection system the features appear much more pronounced now.

![Figure 1](image1.png)

**Figure 1.** Typical profiles of electron temperature and density as obtained for on-axis ECRH heating in the course of the experiments described. Central temperature and density are 1.8 keV and $2 \times 10^{19} \text{ m}^{-3}$ respectively. The shaded areas give the locations of the six radial channels for electron temperature fluctuation measurements. They are located on the high field side of the profile.

![Figure 2](image2.png)

**Figure 2.** Typical cross power spectral density obtained for the four outermost radial channels of the ECE correlation radiometers. The horizontal line corresponds to the statistical significance level (noise level) of the measurement. Central density in this case is $1 \times 10^{19} \text{ m}^{-3}$, the ECRH power is 360 kW.
2.2 Fluctuation Level

The integrated relative temperature fluctuation level is about the same for the two components, though typically slightly higher for the low frequency component. It increases with increasing minor radius. Typical values are 1.2\% and 0.8\% at around 10 cm effective minor radius for the lower and higher frequency ranges and 0.4\% and 0.1\% respectively at the innermost radial position of about 5 cm (Figure 3a and b).

**Figure 3.** Radial dependence of the relative integrated fluctuation level. Figure 3a gives the low frequency part, integrated to 2.5 kHz (components below 500 Hz are suppressed due to mainly technical origin), Figure 3b the high frequency part integrated to 250 kHz.

**Figure 4.** The coherence length as a function of frequency for a 180 kW heated ECRH plasma, showing clearly the difference between the low frequency and high frequency components of the fluctuation spectrum.

2.3 Coherence Length

Besides the 2 times 6 channels in each of the sightlines as discussed above, 12 additional channels are installed mainly in one sightline, allowing radial correlation measurements within this line of sight (and also for fluctuation measurements using the TEXT correlation technique). Figure 4 gives as a typical example the coherence length as a function of frequency for a low power ECRH heated plasma. The coherence length is typically 1-2 cm for the high frequency part of the fluctuations. The low frequency component has a much
longer coherence length of about 4-5 cm. It is found that the coherence length slightly increases with heating power in both the low and high frequency components. At twice the ECRH power of that in Figure 4 the coherence length is 6-7 cm for the range below 10 kHz and 2-3 cm above.

2.4 Heating Power and Heating Scheme Dependencies

While the coherence length slightly increases with heating power, the integrated relative fluctuation level of the high frequency components decreases. A scan of ECRH power under constant density conditions between 200 and 800 kW clearly demonstrates this. In Figure 5 the temperature fluctuations as obtained in the radial range around 5 cm are plotted against the central heating power. In these discharges the electron heat diffusivity clearly increases with heating power P like about \( P^{0.5} \) (power degradation). The findings demonstrate that no simple proportionality between the fluctuation level and the transport properties exist, which could indicate that the fluctuations have nothing at all to do with transport. But transport could still be affected in the expected way if the phase conditions with the necessary poloidal electric field changes or the correlation lengths change (or both) in an unfavourable way. The dependence of the low frequency component on heating power is ambiguous.

![Figure 5. The integrated fluctuation level (only high frequency part) averaged over the effective radius range 5-8 cm as function of on-axis ECRH heating power.](image)

It has been found that the high frequency fluctuation level depends not only on the heating power but also on the heating mode, i.e. O1 or X2 at 70 or 140 GHz respectively. The fluctuation level is smaller under 140 GHz X2 conditions by up to about 30%. This finding was completely unexpected but might have to do with the completely different absorption mechanisms of o- and x-mode and their influence on the electron energy distribution function. O-mode absorption occurs in velocity space at perpendicular velocities around zero but at parallel velocities around 2-3 times thermal velocities, while the conditions for x-mode absorption are the inverse i.e. absorption occurs at perpendicular velocities rather than parallel. In this way the measured fluctuation level might be affected differently by the different heating schemes. This brings up the problem and the possible difference of parallel and perpendicular electron temperatures. ECE radiometry measures the x-component only, therefore reflecting only the temperature fluctuations of the perpendicular component. This results in an asymmetry between the ECRH heating of the plasma and the ECE temperature measurement. Further work is necessary to investigate these most interesting findings.

Due to the heating scheme dependency no electron density dependency will be discussed here. A parameter scan was conducted but mixing different heating scenarios. Separate scans for O1 and X2 conditions will be conducted in near future.
2.5 Phase Velocity

As was discussed before there is a clear difference in the low and high frequency components concerning the coherence length. The most striking difference has been found for the radial phase velocities at frequencies below about 10 kHz and those above. Figures 6a and b give the results. At frequencies below 10 kHz the phase velocity is directed outward, it is of the order of 100 m/s and increases slightly with frequency. At frequencies above 10 kHz the phase velocity is directed inward. It has a much stronger frequency dependence and increases to values of $10^4$ m/s and more. This strong difference in the physical behaviour confirms the idea of completely different mechanisms driving the measured fluctuations.

![Figure 6](image)

Figure 6. The phase velocity as a function of frequency for the different spectral components in the fluctuation spectrum. The low frequency components below 10 kHz are propagating outward (Figure 6a) while those above 10 kHz are propagating inward (Figure 6b). It is supposed that the low frequency components are propagating like a diffusive temperature perturbation. This assumption is confirmed by the phase velocities which are in good agreement with expectations (solid line in Figure 6a).

If we compare the frequency dependence of the low-frequency outward-propagating component with the frequency dependence of the phase velocity of a diffusively outward propagating temperature perturbation, which can be estimated by the slab model expression, $(\omega \chi)^{0.5}$, where $\chi$ is the electron heat diffusivity, which is typically 1 m$^2$/s for W7-AS plasmas under these conditions, and where $\omega$ is the angular frequency component of the perturbation, a very similar behaviour is found (Figure 6a). It could therefore be likely that the low frequency part of the fluctuations is a diffusively outward propagating perturbation probably caused by the central ECRH heating, either direct technically due to a fluctuating power level similar to active perturbation experiments but at much lower level, or due to the electron wave absorption process itself.

3. Origin of 50 kHz Feature

The spectra as shown in Figure 2 and discussed above show a pronounced feature around 50 kHz. No MHD mode activity is observed by other fluctuation diagnostics in these
discharges. We expect to see a turbulent spectrum with a typical \( f^{-\alpha} \) dependence with \( \alpha > 1 \). It is supposed that the feature is caused by Doppler shift due to a poloidal plasma rotation transposing spectral components from near zero into the 50 kHz range of frequencies. Indeed a simple estimation on the basis of measured poloidal rotation velocities (CXRS), which are about 2 km/s in the observation region, results in a Doppler shift of about 50 kHz: \( \Delta \omega = v_{ph} \langle k_\phi \rangle \). The average poloidal k-value, \( \langle k_\phi \rangle \), is estimated to be about 1.5 cm\(^{-1}\). Figure 7a gives the measured poloidal velocity as function of effective radius together with the radial range accessible for temperature fluctuation measurements. Changing the poloidal velocity should change the spectra significantly. To produce these conditions, the plasma aperture was reduced by pushing the limiters to closer positions which shifted the region, of higher poloidal velocity into the observation region, so that now the poloidal velocity is measured to be 12-15 km/s (Figure 7b). As a result, a spectral broadening is observed in all radial channels aside from the inner-most ones where the poloidal velocity is not affected by this measurement. Figure 8 shows the result. It shows that the spectra now extend up to 300 kHz confirming the assumption that the shape of the spectra indeed seems determined by Doppler shift and Doppler broadening.

![Graph showing electron temperature profile with shaded region indicating range accessible to temperature fluctuation measurements.](image)

**Figure 7.** The electron temperature profile with the shaded region indicating the range accessible to temperature fluctuation measurements. Included is the measured poloidal rotation velocity (squares) as measured with CXRS. Limiting the plasma size, the poloidal velocity caused by the ambipolar radial electric field increases in the shaded region from about 2 km/s to about 14 km/s (Figure 7b).
4. Fluctuations at Zero Temperature Gradient

Purely off-axis heating in the W7-AS stellarator results in temperature profiles which are flat over the plasma center. In this way profiles could be established which have no significant temperature gradient over the whole observation region. The ECRH power deposition is centered around 8-10 cm effective radius in this case. The electron temperature in the observation region is comparable to that one obtained in the plasmas characterized by the profiles of Figure 1, which is about 1 keV with a gradient of 180 eV/cm. Because the density is also held constant experimental conditions differ only by the temperature gradient. Figure 9 gives the electron temperature profile and the observation region. To obtain a significantly small statistical significance level, 5 equal stationary discharges each 0.8 s long have been used in this experiment. Figure 10 gives the cross power spectrum for the 4 outermost channels. The most striking observation is that the high frequency components disappear almost completely. The low frequency part is still present with a level which by contrast is somewhat increased. The latter observation confirms the idea of temperature perturbations generated within the heating zone, which under off-axis conditions coincides with the observation region of the correlation radiometers.

Concerning the disappearance of the high frequency components different conclusions can be drawn. If the fluctuations are temperature gradient driven they should disappear under zero gradient conditions. On the other hand, if there is electrostatic exB turbulence which generates the temperature fluctuations, the turbulence must not be affected by the size, the existence or non-existence of a temperature gradient. Nonetheless the temperature fluctuations will completely disappear if there is no temperature gradient present along which turbulence can mix hotter and colder plasma zones. Despite these unsolved problems we think that the disappearance is an important observation which might be very helpful in confirming or excluding theoretical models. At least it is a good experimental test of the temperature fluctuation diagnostic which is not affected at all by high power ECRH heating, despite whose presence extremely small radiation levels are measured with high precision.
Figure 9. Purely off-axis ECRH heating produces flat temperature profiles with gradT_e = 0 in the observation region of the temperature fluctuation diagnostic.

Figure 10. The high frequency part of the fluctuation spectrum almost completely disappears when the temperature gradient is zero in the observation region. The low frequency part is not significantly affected by the different temperature gradient.

5. Correlations of Density and Temperature

A fast, broadband, heterodyne reflectometer [Hartfuss et al 1994, Hirsch et al 1996] which uses two different Gaussian beam antenna systems to launch a wave and to receive the reflected one has been combined with the crossed sightline ECE correlation radiometers to look for the phase relation of density and temperature fluctuations. This is of importance not only from physics reasons, but also from the standpoint that density fluctuations in the case of low optical depth of the plasma can cause fluctuations of the radiation temperature which are then interpreted as temperature fluctuations. This cross-talk would give a systematic error in the temperature fluctuation measurements. First measurements of the combined systems have been conducted. It is found that density and temperature fluctuations are correlated. But there is still no clear picture concerning the phase relation between the two quantities. All relations from in-phase to anti-phase are found depending both on the location and the frequency range of the fluctuations. Thus it is clear that there is not only the in-phase correlation which could come from the cross-talk mentioned above.
6. Summary

With an improved system with regards to poloidal and radial resolution some more details of the temperature fluctuations in purely ECRH heated plasmas of W7-AS have been measured. It is found that the fluctuation spectra consist of clearly separable contributions of different origin. While the low frequency part seems to be mainly a diffusively outward propagating perturbation, propagating away from the power deposition zone with a phase velocity typical for a diffusive process, the high frequency component resembles fluctuations expected for electrostatic turbulence. It disappears almost completely under vanishing temperature gradient conditions and it scales with power, density and rotational transform in absolute level as well as in coherence length. The spectral features of the high frequency component seem to be determined by Doppler broadening and a Doppler shift due to the poloidal rotation of the plasma. The measured temperature fluctuations are correlated with the density fluctuations of the same plasma volume. But a complicated variety of dependencies exists which is still far from understood at the moment. Future work will mainly be dedicated to these parameter dependencies and to a detailed comparison of the two different correlation techniques developed at W7-AS and TEXT.

Concerning the fluctuation driven transport, due to the lack of information on poloidal E-field fluctuations, it is impossible to derive a heat flux from these measurements. Comparison with energy fluxes derived on the basis of dissipative trapped electron theory shows that the measured fluctuation level is comparable to or even higher than the minimum level necessary to explain the measured heat flux, assuming an optimum phase relation and unity correlation between temperature and E-field fluctuations [Wootton and Fonck 1993].

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Edge Ion Temperature Profiles in L- and H-Mode Discharges of ASDEX

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Introduction
A common. and widely believed unalterable, feature of the H-mode is the formation of a steep gradient region of the radial electric field $E_r$ at the transport barrier of the plasma edge. $E_r$ in turn is connected with the ion pressure gradient $\nabla p_i$ and the rotation velocity through the radial force balance equation. Although it is not yet clear what the triggering mechanism for the L-H transition is, the ion temperature respectively its gradient is seen to play a crucial role in the feedback loop of the radial force balance. On ASDEX edge ion temperature profiles have been made accessible with the help of low energy neutral fluxes in an analysis after shutdown. Therefore these profiles are now available for ASDEX L- and H-mode plasmas.

Experimental
Neutral deuterium fluxes in the energy range 15 to 700 eV/amu were measured with the LENA diagnostic [1] at ASDEX with a time resolution of 50 to 100 ms. This neutral (CX) spectrum, which originated mainly from the plasma edge, is simulated with the help of the Monte-Carlo neutral particle code EIRENE. including all available data about the plasma and the geometry near the line of sight. From the fit of the simulated to the experimental spectrum an edge ion temperature ($T_i$) profile of the main plasma species from the separatrix to about 10 cm inside (in the case of ASDEX) can be deduced [2]. This method has been further developed and is now also used on ASDEX-Upgrade [3].

It is known from spectroscopic measurements on ASDEX and other experiments, that the edge $T_i$ increases at the L-H transition. A look at the LENA CX spectra during a H-mode discharge also shows a significant change of shape in different phases. This can be clearly seen in the upper picture of Fig. 1.1: starting from the ohmic phase over the L- to the ELM-free H*-phase the slope of the high energy part increases indicating already a higher $T_i$. At the same time a sharp bend at low energies (100 eV) occurs. The lower picture compares the corresponding ion and electron temperature ($T_e$) profiles and the $T_i$ profile of a comparable L-mode discharge, i.e. where no L-H transition took place. A significant difference between L- and H-phases is found: in the H-Mode, the absolute $T_i$ value and the gradient is much higher than in the L-Mode. The temperature in the H-mode increases from around 50 eV at the separatrix to more than 400 eV within 1-2 cm inside the separatrix, which is even much higher than the electron temperature. The radial range of the steep $T_i$ gradient becomes clearer in Fig. 1.2, where the gradient at the separatrix and some cm inside are compared during the H-mode discharge. From this the transport barrier can be located in a narrow
Fig. 1.1: Low energy neutral spectra and corresponding $T_i$ and $T_e$ profiles in L- and H-phases. If not indicated, the data refer to discharge #33308.

range of 1-2 cm at the separatrix, which is in agreement with similar observations on $\nabla T_i$ and $E_r$ in other machines.

These $T_i$ profiles correct earlier interpretations of the L-H transition at ASDEX [4], where, due to the lack of appropriate edge $T_i$ measurements, the main influence on the transition has been attributed to the edge electron temperature. The $T_i$ profiles in Fig. 1.1 also reveal that already the L-phase prior to the H-transition has a higher edge $T_i$ than in a pure L-Mode discharge. This supports the major role of the ion temperature or its gradient in the L-H transition.

Discussion
One possible interpretation of the L-H transition follows from the obviously different LENA $T_i$ edge profiles: a minimal $T_i$ at the edge seems to be a prerequisite of the H-mode. Therefore
Fig. 1.2: Ion temperature gradient at and inside the separatrix (Spx) with the $H_{\alpha}$-intensity for the identification of the phases in the H-mode discharge #33308.

we have used the $T_i$ edge profiles as an input parameter of the ion orbit loss model of Shaing and Crume [5]. This assumes the loss of collisionless ions across the separatrix as the trigger, which in turn induce the radial electric field. The critical parameter is the ion collisionality $\nu_{\alpha i}$, which forces the L-H transition at values $\nu_{\alpha i} \lesssim 1$. Table 1.1 lists the effective collisionality

$$\nu_{\alpha i}^{eff} = \nu_{\alpha i} \left( \frac{Z_{eff} n_e}{Z_i n_i} \right),$$

(1.1)

where the impurities are taken into account by a $Z_{eff}$ [6], about one poloidal gyroradius inside the separatrix for the L-and H-phases. A typical value of $Z_{eff} = 4$ with an impurity charge of 7 has been assumed. The phenomenological explanation for the pure L-mode discharge is found probably in the double null (DN) configuration (vertical shift $z = 0$), which has a higher power threshold than the single null (SN). It is clearly seen, that in the L-mode #32603 the high value of $\nu_{\alpha i}^{eff} \approx 4$ prevents the L-H transition, whereas in #33308 the
<table>
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<tr>
<th>discharge</th>
<th>time [s]</th>
<th>mode</th>
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<th>$\bar{n}_e$ [cm$^{-3}$]</th>
<th>$T_{i,39}$ [eV]</th>
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<td>1.46</td>
<td>L</td>
<td>2.5</td>
<td>$3 \cdot 10^{13}$</td>
<td>180</td>
<td>1.2</td>
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</tr>
<tr>
<td>#33308</td>
<td>1.07</td>
<td>L</td>
<td>2</td>
<td>$2.8 \cdot 10^{13}$</td>
<td>350</td>
<td>1.9</td>
<td>0.7</td>
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<td>(SN $z = 1$ cm)</td>
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<tr>
<td>1.17</td>
<td>H*</td>
<td></td>
<td></td>
<td></td>
<td>450</td>
<td>2.1</td>
<td>$\pm 0.2$</td>
</tr>
<tr>
<td>1.47</td>
<td>H</td>
<td></td>
<td></td>
<td></td>
<td>380</td>
<td>2.0</td>
<td>0.8</td>
</tr>
<tr>
<td>#33301</td>
<td>1.07</td>
<td>H</td>
<td>2</td>
<td>$3 \cdot 10^{13}$</td>
<td>540</td>
<td>2.0</td>
<td>$\pm 0.25$</td>
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Table 1.1: Effective collisionality $\nu^{eff}_i$ and poloidal gyroradius $\rho_{\Theta,i}$ of the ions for different L- and H-phases. $\nu^{eff}_i$ and $\rho_{\Theta,i}$ are calculated 1 cm inside ($\rho = 39$ cm) the separatrix. The errors in $\nu^{eff}_i$ follow from the $\pm 30\%$ uncertainty in electron density at the edge.

$\nu^{eff}_i = 0.7$ is sufficient for the transition and consequently no change in the H-mode is seen. The long H-phase in #33301 with regular ELMs has due to the low edge density an even smaller $\nu^{eff}_i$. Therefore the L-H transition with respect to our $T_i$ observations is in good agreement with the prediction by the ion orbit loss model.

Conclusions
With the help of the low energy neutral fluxes it has been possible to obtain $T_i$ edge profiles in L- and H-mode plasmas for the first time in ASDEX, the machine, which first discovered the H-mode. High $T_i$ with a large gradient at the separatrix are found in the H-mode, in concurrence with $T_i$ measurements on other experiments. The comparison with the ion orbit loss model of Shaing is in good agreement, however it offers no direct evidence for this explanation.

References
Stellarator optimization studies in W7-AS


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Abstract. Wendelstein 7-AS is the first medium scale modular stellarator, partially optimised with respect to reduced equilibrium currents \( j_{\parallel} \). It, therefore, plays an important role for the optimised next step stellarator W7-X as pilot experiment for manufacturing techniques of coils and vessel and for equilibrium and neo-classical transport properties of the Wendelstein stellarator line of IPP. Experimental results from W7-AS of importance for the continuation and optimisation of the advanced stellarators will be summarised: with respect to magnetic optimisation we will address improved equilibrium, stability and neo-classical transport and the feasibility of the modular coil system. Regarding operational enhancements we will focus on demonstrated high density operation and the soft edge density limit, improved confinement, verification of the 3D plasma edge topology, stability of island structures for exhaust properties and demonstrated stable diverter high recycling in the island divertor configuration.

1. Introduction

Stellarators promise steady state operation without the disruption hazard known from high-current tokamaks. Classical stellarators, however, suffer from specific deficiencies (physics and technical problems), questioning their reactor potential. The physics problems arise from the quality of magnetic surfaces, MHD and neo-classical properties, the technical problems from the complex geometry and the helical coils (and related forces). Due to the generation of the magnetic field in stellarators exclusively by external coils and the true 3D topology of the resulting magnetic field, the geometric and magnetic topologies can be decoupled allowing for optimisation based on physics issues which lead to the so-called advanced stellarator [1]. It has good equilibrium and stability beta, low neo-classical transport and in a future reactor, sufficient \( \alpha \)-particle confinement. Toroidal and helical coils are replaced by modular ones. The optimisation of the stellarator is carried out under the boundary condition of a smooth surface to allow the application of proper exhaust techniques, a low bootstrap current, low global shear, and a magnetic well to provide stability. The optimisation process then yields the details of the plasma shape: aspect ratio, number of periods, shape of the flux surfaces (superposition of different helical components which lead to helical excursion,
ellipticity, triangularity and indentation), and variation of the cross section along the toroidal direction. W7-X will be the first optimised stellarator with respect to all of the above mentioned categories. Its predecessor, W7-AS, in operation at IPP in Garching, is only partially optimised with respect to reduced equilibrium currents. W7-AS, therefore, plays an important role for the optimised next step stellarator W7-X as pilot experiment for manufacturing techniques of coils and vessel and for optimisation properties of the Wendelstein stellarator line of IPP - especially the verification of the predicted improved equilibrium and neo-classical transport issues.

After having introduced the main parameters of W7-AS in chapter 2, we will divide our paper into two major topics, namely magnetic optimisation and operational enhancements. In chapter 3 we will discuss magnetic optimisation characteristics with respect to equilibrium and stability (3.1) and neo-classical transport (3.2). In chapter 4 we will focus on high density operation and density limit issues (4.1), the stellarator confinement database and scaling (ISS95) and on discharges which show improved confinement (4.2) and finally discuss in detail edge plasma and island divertor experiments and modelling (4.3). In chapter 5 we will finally summarise and draw our conclusions.

2. Parameters of W7-AS

W7-AS is a low shear advanced Stellarator with a modular coil assembly of 40 non-planar field coils plus five larger corner coils arranged in a five-fold toroidal symmetry, which resembles a pentagon of five linked mirrors [2], see figure 1. It should be noted that the set of modular field coils alone is sufficient to confine the plasma at a rotational transform of 0.4. An additional set of 10 planar toroidal field coils (not shown in fig. 1) increases the experimental flexibility and allows a variation of the rotational transform in the range 0.27 ≤ \( \tau \) ≤ 0.7. The device is of medium size with major and averaged minor radii of 2 m and 0.18 m, respectively, yielding an aspect ratio of about 10. Small vertical fields are applied to shift the plasma horizontally (e.g. to compensate for a beta induced Shafranov shift), and the OH transformer is commonly used to compensate toroidal currents, such as the bootstrap current, yielding the so-called "net-current-free" operation.

![Figure 1. Top view of the W7-AS modular coil set and plasma. The 5 larger modular coils in the corners of the pentagonal configuration allow to establish mirror fields and, thus, increase the experimental flexibility.](image)

The plasma cross-section varies toroidally between a more triangular shape in between the corners (\( \Phi=0^\circ \)) and an almost elliptical shape at the corners of the pentagon (\( \Phi=\pm36^\circ \)). Via the 5 larger modular corner coils the magnetic field strength can be varied independently, allowing to establish mirror fields, characterised by the mirror ratio \( \text{MR} = (B_{36} - B_0) / (B_{36} + B_0) \). The available heating power in W7-AS has been extended to 0.4 MW ECRH at 70 GHz plus 0.8 MW ECRH at 140 GHz and 3.2 MW NBI. The maximal plasma parameters achieved so far in independent discharges are electron densities of \( n_e \leq 3 \times 10^{20} \text{ m}^{-3} \), electron and ion temperatures of \( T_e(0) \leq 3.5 \text{ keV} \) and \( T_i(0) \leq 1.5 \text{ keV} \), respectively, and confinement times up to \( \tau_e \leq 43 \text{ ms} \). The achieved beta values are \( \langle \beta \rangle = 1.8 \text{ %} \) and \( \beta_0 = 4 \text{ %} \), which are close to the predicted stability limit.

The feasibility of the modular coil set has been demonstrated. The larger corner coils also proved the scaleability of the modular coils to larger units. A high quality magnetic field with smooth magnetic surfaces and a regular edge has been realised in very good agreement with the calculated and expected field geometry.
3. Optimisation of Magnetic System

W7-AS represents just a first step on the way to an optimised stellarator. The goal was to reduce the parallel Pfirsch-Schlüter currents $j_t$ by a factor of about 2. The Pfirsch-Schlüter currents generate a vertical field which gives rise to the Shafranov shift $\Delta_0$ of the plasma axis. The reduction of these currents, therefore, results in an improved equilibrium. Simultaneously, this measure leads to a reduction of neo-classical fluxes in the lmfp-regime, as is mandatory for stellarators.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>W7-A</th>
<th>W7-AS</th>
<th>W7-X</th>
</tr>
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<tbody>
<tr>
<td>aspect ratio</td>
<td>R/a</td>
<td>20</td>
<td>10</td>
</tr>
<tr>
<td>$\iota$</td>
<td>0.05 - 0.6</td>
<td>0.2 - 0.7</td>
<td>0.83 - 1.25</td>
</tr>
<tr>
<td>av. tor. curvature</td>
<td>$C_0/(\tau R)$</td>
<td>1</td>
<td>0.7</td>
</tr>
<tr>
<td>parallel current</td>
<td>$&lt;j_{</td>
<td></td>
<td>^2}&gt;/&lt;j_{\perp^2}&gt;$</td>
</tr>
<tr>
<td>Shafranov shift</td>
<td>$\Delta_0/a$</td>
<td>80 $\beta_0$</td>
<td>20 $\beta_0$</td>
</tr>
<tr>
<td>bootstrap current</td>
<td>$j_{bs, Stell}/j_{bs, Tok}$</td>
<td>$-1$</td>
<td>$-1$</td>
</tr>
<tr>
<td>stability limit</td>
<td>$&lt;\beta_{st}&gt;/%$</td>
<td>0.01</td>
<td>0.02</td>
</tr>
</tbody>
</table>

Table 1. Optimisation characteristics of Wendelstein advanced stellarators W7-AS and W7-X compared to the classical l=2 W7-A stellarator.

Table 1 compares optimisation characteristics of the advanced stellarators W7-AS and W7-X with the classical l=2 stellarator W7-A. The most important parameters for W7-AS are the aspect ratio reduction by a factor of 2 and the reduction of the normalised toroidal curvature $C_0/(\tau R)$. A reduction of the toroidal curvature has been achieved by a proper matching of helicities.

For the classical stellarator, the normalised toroidal curvature is unity and the current ratio is given by $2\lambda^2$, see equation (1).

\[ j_t = \frac{2}{\lambda} \frac{J_L}{\iota} \cos \theta \]

\[ < j_{||^2} > = \left( \frac{2}{\lambda^2} \right)_{cl,stell} \] (1)

\[ < j_{\perp^2} > = \left( \frac{2}{\lambda^2} \right)_{adv,still} \] (2)

\[ \frac{\Delta_0}{a} \approx \frac{R (C_0/(\tau R))}{a (r/R)} < \beta > \] (3)

In case of advanced stellarators the normalised toroidal curvature is smaller than one (2). In the linear approximation, equation (3), the Shafranov shift of the plasma axis is a function of the aspect ratio, the toroidal curvature and iota [3,4]. In case of W7-X the toroidal curvature is further

![Figure 2. Ratio of parallel to perpendicular currents as a function of rotational transform for W7-A, -AS and -X. The strong reduction for the optimised advanced stellarators is compared with the limit $2\lambda^2$ of a classical stellarator.](image-url)
reduced. Together with the increase in iota this results in a very strong reduction of the parallel currents and Shafranov shift. Furthermore, also the bootstrap current in W7-X will be reduced significantly.

In reducing the toroidal curvature from 1 to 0.7 the quadratic dependence on this term yields an improvement by a factor of 2. Together with the factor of 2 in the aspect ratio an improvement in the Shafranov shift by a factor of 4 in the ratio of parallel (Pfirsch-Schlüter) to perpendicular (diamagnetic) currents and the Shafranov shift is predicted. Figure 2 shows calculated data comparing this current ratio for the classical stellarator W7-A, the partially optimised W7-AS and the further optimised W7-X. In addition the limit 2/q² is plotted as a dashed curve. W7-X also gains from the increase in iota, compared to W7-A and -AS.

3.1 Equilibrium and Stability

One main issue in W7-AS is the demonstration of reduced parallel currents and the resulting improvement of equilibrium, stability and neo-classical transport. The reduction of the Pfirsch-Schlüter currents j|| can directly be measured via the associated dipole field [5] or indirectly via the Shafranov shift caused by this field [6]. Figure 3 shows the measured shift, deduced from soft-X ray profiles, compared to the classical stellarator. The measured reduction verifies the predictions and confirms one aspect the concept of the optimisation.

With the extended NBI heating power of 3.2 MW and the improvement of a non-resonant plasma start-up using a 900 MHz high frequency source to provide a target plasma for NBI intensive β studies in a wide range of magnetic fields (0.6 T ≤ B₀ ≤ 2.5 T) have been carried out. Maximum beta values of <β> = 1.8 % and β₀ = 4 % have been achieved for B₀ = 1.25 T, B_z = 21 mT, t_a = 0.43 and nₑ(0) = 2 10²⁰ m⁻³. The increase in plasma energy follows the stepwise increase in NBI heating power up to energies of 17 kJ without indications of a saturation.

Figure 3. Confirmation of predicted factor of two reduction of Shafranov shift by soft-X ray profile measurements as a function of beta.

Figure 4. Comparison of measured and calculated high-β equilibria (<β> = 1.8%, β₀ = 4%). The flux surfaces in the upper part of the picture have been deduced from tomographic reconstruction of soft-X ray emission while the lower part has been calculated by the NEMEC code. The β-profile has been extracted from Thomson scattering. The resulting Shafranov shift of the plasma axis is about 2/3 of the plasma radius.
Therefore, the experimental β-limit seems not to be encountered, yet. No apparent dependence of the maximum \( \langle \beta \rangle \)-values on the magnetic field has been found for \( 0.8 \, T \leq B_0 \leq 1.25 \, T \). Detailed investigations indicate that this behaviour may partly be caused by the degraded heating power at decreasing magnetic field due to increasing orbit losses [7]. At these high beta values the plasma stays in a quiescent state and usually no virulent processes have been observed, indicating the vicinity of the expected stability limit around \( \langle \beta \rangle = 2 \% \) [8].

For these discharges with maximum beta the plasma equilibrium has been deduced from tomographic reconstruction of soft-X ray emission, see figure 4. Pressure profiles have been extracted from Thomson scattering where the ion contribution has been adjusted to reproduce the measured plasma energy content. These measurements, in excellent agreement with finite-β calculations from the NEMEC code, show a strong (but still reduced) shift of the plasma axis by about 12 cm and clearly verify the predicted high-β equilibria.

**Figure 5.** Stability analysis for a high-β equilibrium (\( \langle \beta \rangle = 1.8\%, B_0 = 4\% \)), calculated by NEMEC. The Shafranov shift, seen in the upper part has caused a significant shift of the plasma axis and leads to a strong modification of the iota profile with respect to the flat vacuum case. The shaded area indicates the resistive interchange unstable region.

Next we want to address the MHD stability limit with respect to pressure driven modes. Due to the strong vertical field we have to apply in our high-β discharges, to prevent the plasma from strong interaction with the outboard vessel structures, the stabilising magnetic well is significantly reduced. As a consequence, we find that these high-β discharges are resistive interchange unstable over 1/3 of the outer plasma radius, but ideal interchange (Mercier) stable over the whole plasma [9], see figure 5. Furthermore, we find a strong increase in shear and the appearance of low order rational surfaces (1/3, 1/4) in or close to this unstable region.

Nevertheless, we do not have experimental indications for strong instabilities and we do not observe any degradation in confinement.

In order to test the predictions of stability theory in W7-AS, we have reduced the magnetic well further by introducing a strong negative toroidal ripple of -20% via proper current ratios in the module and corner coils. Theoretical investigation of the stability limits in these discharges predict resistive unstable regions over substantial parts of the plasma even at low \( \langle \beta \rangle \) [10]. However, up to now we have no experimental indications for strongly increased MHD activities or the onset of instabilities.

High-β equilibrium and stability, which rest on magnetic well stabilisation, will be improved for W7-X by increase of the magnetic well and suitable flux surface shaping. Magnetic well stabilisation can be provided by increased helical curvature in combination
with flux surface shaping. Key elements of proper flux surface shaping are a suitable combination of matched helicities, indentation, ellipticity and triangularity, resulting in the magnetic configuration of W7-X. The reduction of $j_{\parallel}$ contributes to high-$\beta$ stability and improves stability with respect to resistive interchange modes. Low shear is maintained to avoid low order resonances in the confinement zone. Such a configuration is very stable against $\beta$-induced changes of the flux surfaces, the t-profile and shear.

Since the bootstrap current has opposite directions in toroidal and helical symmetry it can be minimised by a suitable combination of toroidal and helical components in the magnetic configuration, as has been done for W7-X. This has been shown in detail in ATF [11]. Furthermore, the measured bootstrap currents in W7-AS are in very good agreement with neo-classical calculations [12].

3.2 Neo-classical Transport

Neo-classical transport of standard stellarators at low collisionality $v$ (long mean free path regime, LmfP) is governed by drifts of particles trapped in the local helical field component (helical ripple) giving rise to the $1/v$-transport regime, see triangles in figure 6. Under the influence of the ambipolar radial electric field the ions tend to a tokamak-like dependence (decreasing transport coefficients with decreasing collisionality) but at a higher level. The electron drift surpasses the ion plateau level in the reactor relevant LmfP regime and the overall neo-classical heat conductivity is unacceptably high and improvement is mandatory.

![Figure 6. Normalised mono-energetic transport coefficients $\Gamma_{11}$, at about half the plasma radius versus normalised collisionality $v/v_n$ ($v_n$ = particle velocity) for W7-A, W7-AS, W7-AS optimised and W7-X at different values of the normalised ambipolar radial electric fields $E_r/v_n$: 0 (triangles), $10^{-4}$ (squares), $3 \times 10^{-3}$ (dots) and $10^{-3}$ (diamonds), respectively. The axisymmetric equivalent tokamak case is indicated by the bold line.](image)

A key element for the confinement of trapped particles is symmetry, since the guiding centre particle orbits are solely determined by the field structure. In axial (tokamak) or helical (quasi helical systems) symmetry passing particles are confined and trapped particles perform banana orbits. In the general 3D case trapped particles are not confined. A reduction of radial drifts away from the flux surfaces can, nevertheless, be achieved by proper combination of toroidal and helical curvature and by localisation of trapped particles in the region of low toroidal field variation. Thus, the influence of the magnetic field ripple which causes the radial drifts and the resulting neo-classical transport can be strongly reduced.

In the W7-AS standard configuration the transport coefficients in the PS- and plateau regime, which are dominated by the toroidal curvature and iota, are significantly reduced compared to the classical W7-A, see figure 6, top right and left. However, due to the larger helical ripple the plateau regime has become smaller and transport in the LmfP regime has increased. The $1/v$-dependence is overcome, like in the classical stellarator, by the influence
of the ambipolar electric field. In an partly optimised configuration with adjusted mirror term and reduced ripple trapped particles are shifted into the straight parts of the pentagon (with low toroidal curvature). The resulting plateau regime is broader and the transport in the lmfp regime is reduced by more than a factor of 2, see figure 6, bottom left. Optimisation carried out in W7-X reduces neo-classical transport much further, see figure 6, bottom right [13].

Figure 7. Variation of the magnetic field on axis as a function of the toroidal angle $\Phi$ along one of the five modules (0$\leq$$\Phi$$\leq$72$^\circ$). The field is normalised to the resonant field $B_{res}$ at the ECRH launch position ($\Phi=36^\circ$) at the corner of the pentagon, see also figure 1. The transport reduced configurations Opt15 and OptBz have significantly lower ripple than the standard configuration and trapped particles are partly shifted towards $\Phi=0^\circ$ / 72$^\circ$.

Experimentally, such transport reduced configurations with minimum ripple have been realised by suitable adjustment of the magnetic fields. They can be realised by the application of a vertical field of the order $B_z/B_0 = 0.01$ or by a mirror component with increased currents in the corner coils ($i_5$) compared to the other modular coils ($i_{mod}$) of $i_5/i_{mod} = 1.2$, shown in figure 7. Such configurations yield mirror ratios MR, defined above, of MR ($i_5$) = +1.5% and MR ($i_{mod}$) = -0.5% compared to our standard case with MR (S) = -1.5%. Additionally, in the transport reduced configurations the number of trapped particles in the regions of strong curvature (ECRH resonance region at $\Phi=36^\circ$ in figure 7) is significantly reduced and partly shifted into the nearly straight regions ($\Phi=0^\circ$ / 72$^\circ$ region in figure 7) [14].

Figure 8. Measured and calculated profiles of the ambipolar radial electric field. The full dots are deduced from CXRS measurements of poloidal and toroidal rotation and the diamagnetic contribution from the He$^{2+}$ pressure profile, where He was used as seed impurity. The open squares are data from Langmuir probes. These measurements are in reasonable agreement with neo-classical calculations from DKES.

The ambipolar radial electric fields, which strongly decrease the transport coefficients in the lmfp regime and allow to surpass the $1/v$ transport losses, as calculated from neo-classical theory by the DKES code [15], have been measured and verified in the experiment, see figure 8. By Charge Exchange Recombination Spectroscopy (CXRS) radial profiles of the electric field have been inferred from measurements of radial profiles of poloidal and toroidal rotation and pressure profiles of He$^{2+}$, the selected seed impurity. Outside the separatrix the radial electric field has been measured by Langmuir probes. Comparison with DKES calculations yields good agreement and the formation of the negative radial $E_r$-profile (ion root) follows the neo-classical model [16].
The achievement of high ion temperatures was initiated by the theoretical prediction of reduced transport by an optimised magnetic field configuration. Indeed the highest ion temperatures of $1.5 \text{ keV}$ are achieved in neo-classically optimised configurations and confirm the neo-classical behaviour of ion transport in W7-AS [17]. However, in these low density discharges ($n_e(0) = 5 \times 10^{19} \text{ m}^{-3}$) a very good wall conditioning and recycling control (boronization, He-GDC) has been a further pre-requisite. The high ion temperatures have been achieved by direct ion heating through $1.3 \text{ MW}$ of NBI added to $0.35 \text{ MW}$ of ECRH, necessary for plasma start-up and initial electron heating.

4. Operational Enhancements

In addition to the direct optimisation criteria and their verification by experiment there has been significant progress in the various modes of operation, some of which we will discuss in this chapter.

4.1 High Density Operation and Density Limit

A maximum operational density limit in stellarators is not determined by a disruption like in tokamaks but by a soft and slow decrease of the energy content after a maximum has been reached. The density limit is defined as the density at maximum energy content prior to this slow radiative quench. Due to the very high densities of $n_e = 3 \times 10^{20} \text{ m}^{-3}$ accessible in W7-AS only beam heated plasmas can be investigated (cut-off for $140 \text{ GHz}$ ECRH: $1.2 \times 10^{20} \text{ m}^{-3}$). In combined magnetic field and density limit scans we find a linear scaling of the density limit values with magnetic field, as shown in figure 9, left. Only as a reference, these values exceed those of ASDEX and the respective Greenwald limit ($n_{e,\text{max}} = l_p/\pi a^2$, $l_p$ replaced by $a$ and $B$) by a factor of three. From comparing W7-AS with CHS, there is, however, no evidence of a linear major radius scaling. We also find a very favourable scaling of the density limit values with heating power: $n_{e,\text{max}} \sim P^{0.4}$, see figure 9, right. Furthermore, we find a strong increase of the density at the last closed flux surface from $4 \times 10^{19} \text{ m}^{-3}$ ($B_0 = 0.6 \text{ T}$) up to $1.2 \times 10^{20} \text{ m}^{-3}$ ($B_0 = 2.5 \text{ T}$). The stable ultra-high density operation capability of stellarators offers a great potential for save divertor operation.

![Figure 9. Comparison of density limit scalings in W7-AS and ASDEX as a function of magnetic field (left) and heating power (right) [18].](image)

In NBI power scans ($0.2 \text{ MW} \leq P_{\text{NBI}} \leq 3 \text{ MW}$) we find at low heating power a good agreement with the power balance ($P_{\text{NBI}} = P_{\text{rad}} + W/\tau_E$), indicating a global limit. At high heating power ($P_{\text{NBI}} > 2 \text{ MW}$) a saturation of energy and density seems to occur [19]. Measurements by our video diagnostic, Lithium beam and bolometers indicate in these cases a radiative violation of the power balance in the plasma periphery.
The soft and slow quench of discharges at the density limit followed by a total recovery has been documented by tangential video measurements of the plasma. In discharge #34753 ($B_0 = 0.6 \, T$, $B_z = 7.2 \, mT$, $t_a = 0.43$) we reach a density limit (maximum in plasma energy) of $n_e(0) = 1 \times 10^{20} \, m^{-3}$ at a plasma energy of $W_{\text{dia}} = 3 \, kJ$. After that maximum the plasma energy decreases gradually but slowly at a rate of $1 \, kJ / 50 \, ms$ while the NBI heating power and the line averaged density stay constant. On the other hand the discharge is still fuelled by a strong gas feed of order $3 \times 10^{21} \, \text{particles per second}$. When the energy decreases, the video measurements show a clear contraction of the hot confinement region. After switching off the gas feed the line averaged density still remains constant but the energy rapidly increases to about the previous maximum level and electron density profiles from the LiB exhibit a strong increase of the edge density gradients. The video measurements clearly show the full recovery of the discharge as demonstrated in the again clearly visible edge plasma structure. We interpret these measurements by an increase of the neutral gas recycling up to a level, where the power balance of the plasma periphery is violated by increasing radiation which can no longer be compensated by heat conduction. At that stage the plasma starts to shrink, driven by radiation. With the termination of the gas feed the target recycling is reduced and the discharge recovers. It should be noted that this whole scenario is stable and allows for control and feedback in strong contrast to a disruptive behaviour in tokamaks.

Figure 10. Comparison of ISS95 data and scaling with tokamak L-mode data.

4.2 Improved Confinement

W7-AS confinement results are integrated in the international stellarator data base activities [20] and the analysis has been completed (ISS95 scaling). The database comprises a total of 859 discharges from the high shear heliotrons / torsatrons ATF, CHS and Heliotron E and the shearless stellarators W7-A and W7-AS. It covers ECH and NBI discharges in the L-mode confinement regime. In contrast to tokamaks, the devices fall into two groups distinguished by their magnetic configuration. The heliotron / torsatrons have strong positive magnetic shear and a radially limited magnetic well; the shearless stellarators are characterised by a magnetic well throughout the plasma cross-section. The results of the regression analysis for W7-AS, adopting the major radius dependence from the heliotron/torsatron line, is

$$\tau_E^{W7-AS} = 0.115 \, a^{2.21} \, R^{0.74} \, B^{0.73} \, n^{0.5} \, P^{-0.54} \, t_{2/3}^{0.43}$$

The ISS95 scaling expression is

$$\tau_E^{ISS95} = 0.079 \, a^{2.21} \, R^{0.65} \, B^{0.83} \, n^{0.51} \, P^{-0.59} \, t_{2/3}^{0.4}$$

It should be noted that the confinement of the shearless stellarators exceeds that of the ISS95 scaling by about $35 \%$.

Comparison with tokamaks is a key element, since both concepts share many anomalous confinement issues [21]. In this comparison it is crucial to use the appropriate definitions for minor radius and iota. It is shown that the ISS95 scaling also describes tokamak data.
surprisingly well, see figure 10. Also on the basis of other expressions, the stellarator and tokamak L-mode are of comparable confinement quality [20].

**Figure 11.** Improved confinement in a transport reduced magnetic configuration with confinement enhancement of 1.5/2 above the W7-AS and ISS95- scalings, respectively.

The operational rage for the establishment of the H-mode, which had been found in a narrow operational window of $0.52 \leq \tau_a \leq 0.53$, characterised by the closed 5/9 island chain and magnetic separatrix close to the target, and in ECRH discharges, has been extended. A second operational window at $\tau_a = 0.475$, characterised by the 5/10 island chain and very similar edge topology, as identified by target video, has been established. This confirms the importance of plasma edge conditions, e.g. distinct spatial potential variations at the separatrix, for the development of the H-mode barrier. With NBI H-mode discharges were identified in the iota range $0.52 \leq \tau_a \leq 0.53$ with similar characteristics. In general, H-mode discharges are characterised by fast transitions of edge phenomena on a 100 \( \mu \)s time scale [22].

Besides the H-mode, there exist other confinement modes with confinement times distinctly exceeding the predictions of the ISS95 scaling. As an example we show the temporal evolution of the energy content and confinement time normalised to the W7-AS scaling, $\tau_E / \tau_E^{W7-AS}$, for the discharge #34187 which has a transport reduced configuration, at $\tau_a = 1/3$ and $B = 2.5$ T, figure 11. The density reaches a plateau of $n_e(0) = 1 \times 10^{20}$ m\(^{-3}\) at $T_i = T_e = 0.9$ keV. The confinement time starts around a value comparable to the W7-AS scaling prediction, continues, however, to rise and finally exceeds the W7-AS scaling by 70%. This yields confinement times of about a factor of 2 higher than the ISS95 prediction. It should be noted that this improved confinement has been achieved by an increase of $T_e$ rather than $n_e$, which demonstrates that bifurcations in transport are also possible in stellarators.

### 4.3 Plasma Edge and Island Divertor

**Figure 12.** Calculated island divertor configuration at $\tau_a = 0.510$ characterised by a 5/10 natural island chain at the plasma boundary, intersecting the inboard target, for $\beta=0$ (vacuum field). The intrinsic diversion of power and particle exhaust is provided by particle flows along the island separatrices, which intersect the targets.

The boundary magnetic topology of W7-AS at high $\tau_a$ is characterised by 5/m natural islands (8 $\leq m \leq 12$), see figure 12, and is truly 3D [2]. Therefore, power and particle exhaust in advanced stellarators can only partly benefit from the evolution of the tokamak poloidal divertor. However, the natural edge island topology intrinsically provides the necessary diversion properties, resulting in the so-called island divertor. The basic topological features of this concept for low shear stellarators will be discussed. Extensive measurements of the edge
structures have been performed which will be compared with 2D- and 3D modelling results [23, 24]. In W7-AS crucial elements of this concept have been assessed and these studies have to clarify to what degree boundary islands exist and are stable with respect to equilibrium currents at finite plasma pressure $\beta$. Subsequent investigations focus on the main question whether high recycling and related divertor scenarios can be achieved [25, 26].

The existence of the natural islands, as predicted from the 3D vacuum topology, has been visualized by tangential video, figure 13, right. For a configuration with a 5/8 natural island boundary at $t_{\phi} = 0.6$ the plasma has been viewed in the light of C$^{2+}$ (465 nm). This ionisation stage visualises the plasma boundary around the last closed flux surface (LCFS). Since the video camera views tangentially along the straight part of the torus it integrates the complete three dimensional plasma edge structure in a perspective view and clearly visualises the plasma configuration. Calculations of this perspective view of a single flux surface close to the LCFS toroidally along 56° of the torus clearly reproduce all the features seen in the video pictures, figure 13, left, verifying the existence of the 3D topology and its agreement with the vacuum field structure at low $\beta$.

Figure 13. Calculated (left) and measured (right) tangential video image for a configuration with 5/8 boundary islands at $t_{\phi} = 0.61$, along 3/4 of a module. All the features in the measured image are reproduced in the calculated vacuum field case, verifying the full 3D topology.

In a further step the interaction of the islands with the inboard divertor targets has been investigated in detail. The 10 periodic targets are CFC graphite blocks of 12 cm width and 23 cm height, and thus rather small i.e. they cover only a very small toroidal area, but preserve the five fold symmetry of the machine. In W7-AS, the configuration with the 5/9 islands at the edge offers a good compromise between large plasma and large island size, and has therefore been chosen for standard divertor operation. With such an closed island chain in front of the targets, a slight decrease of the edge rotational transform shifts the islands chain outward, eventually intersecting the targets. Recycling and power deposition at the targets as well as the edge plasma structure have been investigated by H$_{\alpha}$-arrays, target video, thermography, calorimetry and Langmuir probe arrays throughout the whole accessible iota range [23, 26-28]. The measured plasma edge structure and the interaction of the islands as they approach and finally intersect the targets is found to be in excellent agreement with predictions and congruent in between all the used diagnostics. Good consistency is achieved in the mapping of the applied diagnostics, distributed around the torus at topologically different positions, onto flux surfaces.

If the islands are intersected by the targets, divertor fans are formed along the outer island separatrices. With respect to the stability of these structures against moderate $\beta$ we have compared measurements of the target videos and calculations for vacuum and $B_0 = 1\%$ with the KW equilibrium code [29]. The calculations show an increased radial elongation of the islands due to the $\beta$-induced equilibrium currents. The most relevant change with $\beta$, however, is the doubling of the field-line pitch inside the islands, which reduces the connection length to half its value. The island surfaces remain intact and their phase unchanged. This is
confirmed by the target video pictures, showing the island strike points via $H_\alpha$-emission of recycling particles. For both configurations, the locations of the observed $H_\alpha$ stripes on the target coincide very accurately with the predicted strike points, proving the reliability of the calculated finite $\beta$ equilibrium edge configuration. The structures and phase of the observed $H_\alpha$ stripes, which resemble the footprints of the island divertor fans on the targets, are found to be stable up to our highest beta values of $\beta_0 = 4 \%$, corresponding to $\langle \beta \rangle = 1.8 \%$. This demonstrates the robustness of the 5/9 islands for divertor relevant applications [23].

Having confirmed the topological structure of the edge plasma and the stability of the islands against $\beta$-induced equilibrium currents we will now discuss the achievement of divertor high recycling. For line averaged densities $0.2 \times 10^{20} m^{-3} \leq \bar{n}_e \leq 1.5 \times 10^{20} m^{-3}$ and heating powers of 0.8 MW for $\bar{n}_e \leq 0.8 \times 10^{20} m^{-3}$ and 2 MW for higher densities, net-current-free NBI discharges at $B = 2.5$ T, $\tau_B = 0.564$, corresponding to a 5/9 island boundary configuration, have been analysed. Typical island dimensions in the radial and poloidal directions are 5-10 cm, depending on the helical position. Edge plasma parameters were obtained from two Langmuir probes (CFC tips): a fast reciprocating probe (FRLP) at an upstream position with a strike line through an island, and a second probe close to a target. $n_e$ and $T_e$ profiles were derived by fitting the probe characteristics up to the floating potential. The measurements were completed by Thomson scattering, spectroscopic observations ($H_\alpha$ diode arrays viewing the targets, CCD cameras with $H_\alpha$ and C$^2+$ filters), bolometry, low-energy CX neutral analysis (LENA) and target thermography [24, 30].

The edge plasma was modelled in 2D by the B2-EIRENE code [31-34] and in 3D by the newly developed EMC3-EIRENE [23, 35] code. The B2-EIRENE approach allows to study basic divertor properties with sophisticated physics, but includes helical averaging of the island configuration and can in particular not treat the present target geometry. Furthermore, it does not allow direct reference to local experimental data in the actual 3D configuration. Basically, it treats a single island configuration, corresponding to a tokamak single null case. The EMC3 code is fully self consistent in magnetic co-ordinates and considers the actual 3D geometry plus targets, but includes at present some physics simplifications like a single fluid plasma, neglect of heat convection and a parametrisation of momentum losses from experimental data.

![Figure 14. $n_e$ profiles from a Langmuir probe across a 5/9 boundary island at different line-averaged densities and the respective EMC3/EIRENE results (lines). At an averaged density of $1.2 \times 10^{20} m^{-3}$ the profile peaks inside the island close to the outer separatrix, indicating high recycling. The inward shift of this profile peak at higher densities indicates partial detachment. The resulting $n_e$ profiles across a 5/9 boundary island measured by a Langmuir probe at different line-averaged densities are shown in figure 14. Starting at the left we have the main plasma, followed by the island region of about 5 cm width and the private region to the right. The profiles show flattening inside the island at low density, and stronger than linear increase and peaking with increasing averaged density. This gives, together with strongly increasing downstream densities evidence for the establishment of divertor high recycling.](image_url)
At the highest density, the peak becomes inward shifted which indicates at least partial detachment, a scenario which will be studied in more detail in the near future.

The lines indicate respective EMC-3-EIRENE results, which have been matched to the experiments at only two radial points and a downstream probe position close to the target, reproduce the whole profiles quite well.

Upstream and downstream densities and temperatures from the code and probe results versus the averaged density demonstrate increasing parallel temperature gradients and strongly increasing downstream density with increasing averaged density. At highest density, the downstream density shows "roll over" indicating detachment as was already mentioned [24].

In summary, the measurements and calculations clearly show that we can establish stable divertor high recycling in this geometry. Detachment is indicated at highest density, but has to be further studied.

The conditions for high recycling and related divertor scenarios can be substantially improved by the implementation of a new divertor in W7-AS [36].

5. Summary

W7-AS is a partly optimised advanced stellarator with respect to reduced parallel currents, improved equilibrium and neo-classical transport. Additionally, it is the first stellarator to demonstrate the feasibility of modular coils and the resulting high quality magnetic fields. The next step device W7-X will be further optimised.

The predicted magnetic optimisation, which is based on a reduced mean toroidal curvature, matched helicities and proper flux surface shaping, has been studied intensively and has been verified by experimental measurements. The reduction of the Shafranov shift by a factor of two with respect to an equivalent classical stellarator has been demonstrated. High-\(\beta\) plasmas up to \(|\beta| = 1.8\%\) and \(\beta_0 = 4\%\) have been achieved. The calculations of these equilibria are in excellent agreement with flux surfaces deduced from soft X-ray tomography and \(\beta\)-profiles from Thomson scattering. An experimental \(\beta\)-limit has not been reached so far. Despite the fact, that the achieved high-\(\beta\) values are very close to the predicted stability limit of \(|\beta| = 2\%\), no indications of strong instabilities have been observed up to now.

Neo-classical transport has been investigated in transport reduced configurations, realised via a reduction and shift of trapped particles into regions of low curvature by means of vertical and / or mirror field components. The neo-classically predicted ambipolar radial electric field, which is necessary to overcome the 1/v behaviour in the Lmfp-regime, has been verified by CXRS. In such transport reduced configurations the highest ion temperatures of 1.5 keV have been reached and neo-classical ion transport in the Lmfp has been demonstrated.

In addition to the verification of direct optimisation criteria, there has been significant operational progress. In high density operation density limits of \(\bar{n}_e = 2 \times 10^{20}\) m\(^{-3}\) have been achieved at B = 1.20 T. A linear density limit scaling with magnetic field strength and a favourable P\(^0.4\) scaling with heating power have been found. The density limit, in contrast to the disruptive limit in tokamaks, is found to be very soft and slow so that stable operation at very high densities is possible. The quality of stellarator confinement is comparable to that of the tokamak L-mode, which are both very well described by the international stellarator scaling ISS95. Confinement in W7-AS, however, exceeds ISS95 by about 35% and in transport reduced configurations improved confinement with an enhancement factor of 2 with respect to ISS95 has been found. Plasma edge and island divertor topologies have been studied very intensively in W7-AS. The existence of the natural islands and the 3D-edge topology at low and high \(\beta\) has been demonstrated by a large set of diagnostics, in very good agreement with code calculations. Stable divertor high recycling has been established
and has been successfully modelled by the new three-dimensional EMC3-EIRENE code package. These conditions for high recycling and divertor operation will be substantially improved by the implementation of a new divertor.

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Stellarator Optimization Studies in W7-AS

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Parameters of Wendelstein 7-AS Stellarator

- Modular Design
- 5 Periods
- Low Shear

Optimization Characteristics

- Magnetic Optimization
  - Equilibrium and Stability
  - Neoclassical Transport (LMFP)

- Operational Enhancements
  - Density Limit
  - Good Confinement
  - Plasma Edge, Island Divertor

Optimization Characteristics

Parameter: R/a

- W7-A: 20
- W7-AS: 10
- W7-X: 10

Parameter: iota

- W7-A: 0.5
- W7-AS: 0.5
- W7-X: 1

Parameter: C_{01}/(r/R)

- W7-A: 1
- W7-AS: 0.7
- W7-X: 0.4

Parameter: <j_i^2>/<j_i^2>

- W7-A: 12
- W7-AS: 3
- W7-X: 0.5

Parameter: Δα/a

- W7-A: 80 β_0
- W7-AS: 20 β_0
- W7-X: 1.6 β_0

Parameter: j_{Stell}/j_{StTok}

- W7-A: ~1
- W7-AS: ~1
- W7-X: 0.1

Parameter: β_{β_{<θ>}}

- W7-A: 0.01
- W7-AS: 0.02
- W7-X: 0.05

W7-A

- Classical l=2

W7-AS

- Part. Optimized

W7-X

- Optimized

Calculated Reduction of Parallel Currents (PS-currents)
in Advanced Stellarators

- Shafranov Shift: Linear Approximation:
  \[ \Delta = \frac{R}{a} \left( \frac{C_{01}}{r/R} \right) \]
  - Aspect Ratio R/a
  - Toroidal curvature C_{01}
  - iota
Confirmation of Predicted Reduced Shafranov Shift
from Soft-X ray profile measurements

W7-AS (iota < 0.4)

W7-AS (iota > 0.4)

Verification of High-β Equilibrium
excellent agreement between theory and measurement for \( \langle \beta \rangle = 1.8\% \), \( \beta_0 = 4\% \)

- Experiment
  - soft-X Tomographic reconstruction
  - \( \beta \) profile from Thomson scat.
  - \( B_0 = 1.25T \), \( B_0 = 32mT \), \( \beta_0 = 0.38 \),
    \( T_0(0) = 2 \times 10^{19} m^{-3} \), \( T_e(0) = 0.35 \) keV

- Theory
  - NEMEC finite-\( \beta \) equilibrium
    \( \Rightarrow \langle \beta \rangle = 1.8\% \), \( \beta_0 = 4\% \)
  - Good agreement: measured and calculated flux surfaces shift of plasma axis!

High-β Plasmas with Extended NBI-Heating
achieved maximum values: \( \langle \beta \rangle = 1.8\% \), \( \beta_0 = 4\% \)

- Extended NBI-Heating with 3.2 MW
  - 2.4 MW absorbed NBI power
  - B-scan: \( 0.6T \leq B_0 \leq 2.5T \)

- Achievement of High \( \beta \)-Values
  - \( B_0 = 1.25T \), \( B_0 = 21mT \),
    \( \Rightarrow \langle \beta \rangle = 1.8\% \), \( \beta_0 = 4\% \)
  - No saturation in energy content
  - No apparent \( B_0 \) dependence (0.8T - 1.2T)
  - Quiescent plasma behaviour close to stability limit, no occurrence of violent processes

High-\( \beta \) discharge

Investigation of MHD Stability Limit
no instabilities experimentally observed

- \( \beta \)-Limit
  - Encounter of experimental \( \beta \)-limit not yet clear

- High-\( \beta \) Configurations
  - Reduced magnetic well, \( \beta_0 = 4\% \)
    resistive interchange unstable, Mercier stable
  - Strong increase of shear and appearance of low order rational surfaces

- More Unstable Configurations
  - \( \beta \)-20% neg. tor. ripple
    resistive interchange unstable at low \( \langle \beta \rangle \), Mercier stable;
    similar MHD activities as standard configuration!

\( \Rightarrow \) no instabilities experimentally observed!
Neoclassical Transport Regimes

- Helical Ripple Losses Dominate LMFP Regime: 1/4-regime
- Shift of trapped particles into good curvature region mirror component
- Tor. Curvature and Iota Dominate PS and Plasma Plateau Regime

Normalized transport coefficients at r/a=1/2

- W7-A
- W7-AS
- W7-X

Optimization of Neoclassical Transport in W7-AS
by vertical field and mirror component

- Optimized Configuration
  - Minimum ripple by optimized magnetic field
  - Vertical field \( B_z/B_0 = 0.01 \), inward shift of plasma
  - Mirror field \( \lambda_{min} = 1.2 \), mirror component
  - Reduction + shift of trapped particles
  - Reduction of heat conductivity
  - Mirror ratio: \( MR = (B_{36}-B_0)/(B_{36}+B_0) \)

Verification of Radial Electric Field
by CXRS and probes in agreement with neoclassical theory

- Predicted Ambipolar \( E_r \)
  - Neoclassical calculations (DKES) predict improved transport for moderate \( E_r \) in LMFP

- Verification of Neoclassical Ambipolar \( E_r \)
  - CXRS: pol. + tor. rotation and He+++ pressure profile
  - Langmuir probes
  - Radial profile of \( E_r \) (ion root)
  - Formation of \( E_r \) follows neoclassical model

High Ion Temperatures in Optimized Configuration
transport optimized by vertical field: \( T_i(0) = 1.5 \text{ keV} \)

- Transport Reduced Configuration
  - Minimum ripple by vertical field \( B_z/B_0 = 0.01 \), \( B_{pol} = 10 \)
  - Predicted transport improvement by a factor of 2 (DKES)
  - Predicted \( E_r \) confirmed by CXRS
    - \( E_r = 400 \text{ V/cm}, v_i/2 \text{ regime} \)
  - Direct ion heating by NBI (1.3 MW)
  - Low density by wall conditioning
  - Achieved \( T_i(0) = 1.5 \text{ keV} \)
High Density Operation, Density Limit
- slow radiative limit, factor 3 above Greenwald limit, edge density limit

- Operation at High Density
  - Experiment: 0.6 T ≤ B₀ ≤ 2.5 T and density ($n_{max}$) scan
  - Density limit values by a factor of 3 above Greenwald limit

- NBI Power Scan 0.2 ≤ P_{NBI} ≤ 3 MW
  - Good agreement with power balance at low power: global limit
  - Saturation of energy + density at high power: local edge density limit

Density Limit Scalings
- scaling with respect to magnetic field and heating power

- Density Limit
  - Density at max. energy content prior to slow radiative plasma collapse

- Field Dependence of Density Limit
  - Favourable linear B-dependence observed

- Heating Power Dependence
  - Favourable scaling with heating power

Indication for Edge Density Limit
- soft limit at B₀ = 0.6 T, P₀ = 3.8 %, control via gasflux, recovery after quench

- High recycling + radiation at targets ⇒ slow soft quench
- Recovery after gas is switched off

Good Confinement: Factor 2 Above L-Mode
- stellarator scaling, new confinement regime

- International Stellarator Database ISS95
  - Database of 859 discharges from ATF, CHS, HéE (high shear) and W7-A and W7-AS (low shear)
    - $t_{qISS95} = 0.072 ± 0.02$ G.65 G.85 G.65 p.0.05 ± 0.04
    - $t_{qW7AS} = 0.115 ± 0.02$ G.74 G.85 G.85 p.0.05 ± 0.04
  - Comparable quality of stellarator and tokamak L-mode confinement
  - ISS95 scaling also describes tokamaks very well

- New Improved Confinement ($t_q ≤ 43$ ms)
  - Transport reduced magnetic configuration with high density + low heating power $n_0 = 10^{20}$ m⁻³, $T_0(0) = 0.9$ keV
  - $t_q / t_{qW7AS} < 1.5$, $t_q / t_{qISS95} < 2.5$
Plasma Edge Topology and Targets

5/m natural islands, periodic inboard targets, island divertor geometry

- 2D Edge Topology
  - 5 field periods \(\Rightarrow\) 5/m resonances
  - natural magnetic islands
  - \(\Rightarrow\) favourable diversion properties of natural islands

- Targets
  - 10 periodic inboard targets
  - CFC graphite blocks of 12 cm width, 22 cm height

- Island Divertor Geometry
  - Flux diversion inside magnetic island
  - basic topology as tokamak divertor different geometry

Verification of 3D Edge Topology and Island Signatures

3D plasma structure (5/8 configuration) from video

- Calculated Perspective View of a Flux Surface (~ I.CFS)
  - 5/8 res., \(r_g = 0.6\), \(r_e = 15\) cm, \(-20^\circ \leq \Phi \leq +36^\circ\)
  - Transformation into density distribution

Energy and Particle Deposition at Targets

\(<\beta> = 1\%\), deposition profiles in excellent agreement with field topology

- Stability of 5/9 Island Structure Against Moderate \(\beta\)
  - island signatures at targets, stability of strike points

- Vacuum + Finite-\(\beta\) Equilibria (KW-code)
Transport Modelling for Island Edge Plasma
high recycling: 2D (B2-EIRENE), 3D (EMC3+EIRENE) transport code results

- Modelling Results
- EMC3: fairly well description of local experimental data
- B2: 2D reference solutions

Up-/Downstream Data and Code Results
high recycling and flux enhancement found in data and modelling

- Up-/Downstream $n_e + T_e$
  - Parallel gradients: increase for $T_e$
temperature drop to 10 eV
  - Downstream density: strong increase
  - Data reproduced by code

- Flux Enhancement
  - Increase up to $\approx 35$

- Message
  - stable divertor high recycling established! detachment indicated at highest densities

High Recycling in Magnetic Islands
comparison of probe measurements and 3D modelling at $\beta_0 = 1\%$

- Density Measurements Across an Island
  - 5/9 divertor configuration
  - $n_e$ profiles across an island high recycling, detachment
  - 3D modelling well reproduces measurements
    (EMC3 - EIRENE)
  - divertor high recycling

New W7-AS Divertor, Targets and Baffles
5/9 island configuration, periodic optimized targets

- Optimized With Respect To
  - Position: largest island thickness along helical edge
  - Load distribution: homogeneous wetted areas
  - Recycling: ionization focussed inside island
  - Pumping: neural compression
Conclusions

stellarator optimization studies in W7-AS

- Magnetic Optimization
  - Equilibrium + Stability: reduced Shafranov shift verified high-β equilibria demonstrated no evidence of instabilities near limit

- Neoclassical Transport: transport reduced contig. established ambipolar radial electric field verified high ion temperatures achieved

⇒ predicted optimization verified by experiments

- Operational Enhancements
  - Density Limit: soft limit, very high densities linear βl, favourable Pheat-scaling edge density limit
  - Confinement: stellarator database (tokamak L-mode) new improved confinement by a factor 2
  - Edge + Island Divertor: verification of 3D topology and islands demonstrated β-stability of target deposition establishment of stable divertor high recycling

⇒ extended operational parameter range

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Edge turbulence and transport barrier associated with the H-mode in the W7-AS stellarator

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H-mode operation in W7-AS

In the modular stellarator W7-AS (R0 = 2.0m, a ≤ 0.18m) H-mode operation is achieved in a net currentless plasma with either ECRH or NBI heating at B_{tor} =2.5T and B_{tor} =1.25T (1). The operational range is characterized by narrow windows of the edge rotational transform τ_e (around τ_e =0.525 and τ_e =0.475 ) where the plasma minor radius is comparatively large and determined by the inner separatrix of a natural island chain. For this well defined plasma boundary the connection length decreases to a value of some meters within a radial distance of Δr=1cm outside the LCFS. As a consequence a strong radial variation of the radial electric field and the concomitant velocity shear layer exist already under L-mode conditions. This is considered as a favourable preconditioning for the L-H transition(1): At τ_e =0.525 the H-mode is achieved already at the lowest available heating power, 200kW of ECRH (one gyrotron) or 340kW of NBI (one source) respectively. At the L-H transition the poloidal impurity rotation measured with spectroscopy increases towards the electron diamagnetic flow direction corresponding to a change of the negative radial electric field by ΔE_{rad} ≈ -100V/cm. In analogy to tokamaks the transition into the H-phase is characterized by the appearance of an edge transport barrier. However the improvement of global confinement is comparatively small (ΔW_{pol} / W_{pol} < 30% ) and achieved only after a quiescent H-mode of sufficient length (>30ms) is established. Here we concentrate on the temporal and spatial behaviour of the edge transport barrier and the associated changes of fluctuations in density and magnetic field.

Edge transport barrier in the L- and H-phase

The existence of a barrier for particle and heat transport in the H-phase is confirmed by various diagnostics: In the H-phase the density gradient around and inside the separatrix increases as measured with Thomson scattering, Li-beam, reflectometry and Langmuir probes indicating an improvement of particle confinement. Electron temperature profiles obtained from ECE show the build-up of an edge pedestal inside the LCFS. The corresponding results obtained with several SX-channels are shown in Fig.1. The increase of the edge ion temperature is determined spectroscopically from the broadening of the impurity lines. Similar results are obtained with the LENA diagnostic. The transport barrier is characterized by the so-
called pivot-point inside of which density and temperature start to increase after the transition whereas they decreases outside.

In most cases the H-mode is reached through a "dithering" phase characterized by an almost periodic modulation of the transport barrier indicated e.g. by the $H_{\alpha}$ signal originating outside the LCFS. A typical repetition frequency is 2 kHz. The transient phase of deteriorated energy confinement can be clearly identified by ECE and SX-diagnostics. The modulation of $T_e$ is only noticable in the ECE channels up to 5 cm inside the separatrix demonstrating the edge localized character of the phenomenon. The concomitant changes of the density gradient inside the separatrix are seen with the reflectometer.

**Edge turbulence at the L-H transition**

Fluctuation diagnostics show that plasma turbulence is strongly suppressed in the H-mode: The density fluctuations in the gradient region and around the separatrix are measured by reflectometry and microwave scattering. For radial positions inside the separatrix the spectral power of broadband density turbulence is reduced in the H-mode by more than one order of magnitude. Magnetic fluctuations observed with Mirnov coils (frequency range $f \leq 600$ kHz) mounted close to the vessel wall are reduced on a similar scale. This reduction is observed for all available poloidal and toroidal coil positions. As an example Fig. 2 shows turbulent signal power for two coils at toroidal positions separated by 174°. The magnetic field fluctuations are correlated within $\Delta t < 3 \mu s$.

The dithering L-H transition is characterized by periodic burst-like phenomena lasting 100 - 200 $\mu$s each with typical repetition frequency around 2 kHz. In the short intervals between the bursts fluctuation power of both density and magnetic turbulence is strongly reduced and the frequency spectra are identical to those in the fully developed H-mode. Therefore these quiescent time windows ($\approx 300 \mu$s) are interpreted as short phases where the plasma edge is in an H-state.

If a dithering burst appears the spectral power of the broadband density fluctuations increases by up to two orders of magnitude within several 10 microseconds. This high measured fluctuation level is almost independent of the radial penetration depth of the reflectometer. In contrast in a stationary L-mode the measured fluctuation level typically decreases if the nominal cut-off layer is shifted deeper into the plasma. Together with the bursts in density turbulence bursts of incoherent magnetic broadband fluctuations are observed. In addition to the broadband turbulence the spectra of magnetic fluctuations show other types of instability (e.g. global Alfvén modes). In particular a quasi coherent precursor activity ($f = 400$ kHz) is observed starting 50 $\mu$s before the well defined onset of the density turbulence. $H_{\alpha}$ emission in the SOL begins to rise within $\approx 30 \mu$s with the bursts of density fluctuations demonstrating the fast deterioration of the transport barrier.
At the drop in turbulence the improvement of the transport barrier measured with ECE, SX and reflectometry is found to occur on a timescale of \( \approx 100 \mu s \) (Fig.3). As a burst appears edge electron temperature displayed by the SX channel in Fig.3 increases within a few ten microseconds as a result of the rapid outflux of heat as the transport barrier is disturbed. Thus in the plasma edge region no clear separation in time between the changes in turbulence \( \tilde{n}_e, \tilde{B} \) and in plasma background parameters \( n_e, T_e \) can be found.

It is proposed that the dithering phase does not constitute a simple switching between two states corresponding to equilibrium L- and H- phases: The precursor activity and the radial extent of the level of density turbulence several cm inside the separatrix indicate that the dithering bursts are similar to the high-frequency ("grassy") ELMs seen in tokamaks(2).

**Edge localized Modes**

Typically after a quiescent H-phase of several tenths of millisecond length, single or quasi-periodic ELMs (typical repetition rates 50 and 100 Hz) appear. The duration of these phenomena is \( \tau \lesssim 0.5 \text{ms} \). Broadband density and magnetic turbulence increases far above L-mode level. No clear coherent precursor activity has been identified in the fluctuations of density and magnetic field. With the multichannel SX-camera the \( T_e \) profiles are found to flatten initially around a pivot point about 2.5 cm inside the separatrix. This position corresponds to the maximum of the pressure gradient before the ELM. The breakdown of the transport barrier manifests itself in ECE as sudden decrease of \( T_e \) in the channel \( \approx 2 \text{cm} \) inside the separatrix. Correspondingly a cold pulse can be followed until \( \approx 5 \text{cm} \) inside the separatrix (\( v_r \approx 25 \text{m/s} \)). In some cases an ELM triggers an L-phase of several ms duration ("compound ELM") characterized by both a level of broadband turbulence and of \( \text{H}_\alpha \) emission comparable to stationary L-mode. During this phase the \( T_e \)-profiles measured with ECE remain flattened until the fluctuations return to H-mode level.

**Conclusions**

The H-mode in W7-AS is characterized by an edge transport barrier displaying a fast switching during transient phenomena like dithering ("grassy") ELMs at the L-H transition and ELMs after a quiescent H-phase. Macroscopic profiles and microscopic fluctuations simultaneously change on a timescale of 100\( \mu \)s indicating an interdependence which looks like a closed loop process. The similarity between these edge phenomena in W7-AS stellarator and the findings in tokamaks underlines the generic behaviour of the H-mode transition as an edge phenomenon in toroidal devices.

**References**

Fig. 1: Development of SX-signal around the pivot point (r_{eff}=14cm) during the L-H transition.

Fig. 2: Synchronous temporal behaviour of magnetic signal power during a dithering L-H transition: Mirnov coil #1 and #2 are separated by 174° in toroidal direction. After a quiescent phase single ELMs occur.

Fig. 3: Dithering phase: H_{alpha}-monitor and SX-channel (outside transport barrier) indicate the quasi-periodic build-up and detoriation of the barrier. The corresponding steepening and flattening of the density gradient is measured from the time-delay signal of reflectometry. As an example for turbulent edge activity the fluctuation power of a Mirnov signal is given. The short intervals of high fluctuation activity correspond to the detoriation of the barrier.
The O-X-B mode conversion process was proposed [1] in 1973 as a possibility to overcome the density limit for electron cyclotron resonance heating (ECRH). Here O, X, and B represent the ordinary, extraordinary and electron Bernstein mode. The essential part of this scheme is the conversion of the O-wave launched by an antenna from the low field side into an X-wave at the O-wave cut-off layer. This mode conversion requires an O-wave oblique launch near an optimal angle.

As shown in Fig. 1 the transverse refractive indices $N_x$ of the O-wave and X-wave are connected at the optimal launch angle with a corresponding longitudinal (parallel $B_0$) index $N_{z,\text{opt}} = \sqrt{Y/(Y+1)}$ with $Y = \omega_{ce}/\omega$ ($\omega$ is the wave frequency, $\omega_{ce}$ is the electron cyclotron frequency) without passing a region of evanescence ($N_x^2 < 0$). For non-optimal launch an evanescent region always exists near the cut-off surface. The geometrical size of this evanescent region depends on the density scale length $L = n_e/\langle \partial n_e/\partial x \rangle$, and a considerable fraction of the energy flux can be transmitted through this region, if $L$ becomes small. The power transmission function $T(N_y, N_z)$ is [2]

$$T(N_y, N_z) = \exp\left[ -\pi k_0 L \sqrt{\frac{Y}{2}} \left( 1 + Y \right) \left( N_{z,\text{opt}} - N_z \right)^2 + N_y^2 \right],$$

where $N_y$ and $N_z$ are the poloidal and longitudinal components of the vacuum refractive index and $k_0$ the wave number. This angular dependence ($N_z$-dependence) was used in the experiments to identify the O-X-conversion process. After the O-X-conversion the X-wave propagates then back to the upper hybrid resonance (UHR) layer where the refractive index of the X-wave is connected to that of the electron Bernstein waves (EBW) as shown in Fig. 1 and a complete conversion into EBW's may take place. The EBW's propagate then towards the plasma centre where they are absorbed near the electron cyclotron resonance layer or in the nonresonant case by collisional multiple pass damping.

In our calculations, additionally, we take into account that in a real plasma the conversion layer is not a smooth surface but is due to density fluctuations rough and wavy. This introduces a beam divergence much higher than the intrinsic one and can reduce the O-X-conversion considerably.
With a statistic description of the poloidal cut-off surface roughness (toroidal fluctuations were neglected), the probability density function of the poloidal component $N_y$ (similar to a poloidal beam divergence)

$$p(N_y) = \frac{\lambda_y}{\sqrt{2\pi} \sigma_y} \exp \left( -\frac{N_y^2 \lambda_y^2}{(1-N_y^2)^2 \sigma_y^2} \right) \left( 1-N_y^2 \right)^{-\frac{3}{2}}$$

could be calculated as a function of the fluctuation amplitude standard deviation $\sigma_y = \bar{n}_e/n_e$ ($\bar{n}_e/n_e$ is the relative fluctuation amplitude) and the poloidal correlation length $\lambda_y$. The modified power transmission function $T_{mod}$ (O-X conversion efficiency) is then $T_{mod}(N_z) = \int T(N_y,N_z) p(N_y) dN_y$.

In Fig. 2 the modified transmission is calculated as a function of the parameter $k_0 L$ for various relative density fluctuation amplitudes. In all calculations the poloidal correlation length was assumed to be 2 cm. It can be clearly seen that a significant heating efficiency is obtained only at a very small density scale length or a very low fluctuation amplitude. The flexibility of W7-AS allows to investigate both extreme cases, i.e. target plasmas with $k_0 L \leq 10$ and with a relative density fluctuation amplitude of more than 20% or peaked density profiles ($k_0 L = 60$) with a very low relative fluctuation amplitude of less than 2%. For both cases, high conversion efficiencies were experimentally measured and O-X-B mode conversion for plasma heating could be clearly shown for the first time.

Two 70 GHz beams were launched into a neutral beam (NBI) sustained target plasma at resonant (1.25 T) and nonresonant (1.75 T) magnetic fields. The launch angle of the incident O-mode polarised wave was varied at fixed heating power (220 kW). An example of the nonresonant case is shown in Fig. 3. The increase of the total stored plasma energy (from the diamagnetic signal) depends strongly on the launch angle, which is typical for the O-X-conversion process, and fits well to the calculation. Here the power transmission function was normalised to the maximum energy increase. The central density was 1.5 $10^{20}$ m$^{-3}$, which is more than twice the cut-off density, the central electron temperature was 500 eV. Heating at the plasma edge could be excluded since at the nonresonant magnetic field of 1.75T no electron cyclotron resonance exist inside the plasma. Due to technical limitation of the maximum launch angle, only the left part of the reduced transmission function could be proved experimentally.

In the X-B-conversion process near the UHR parametric instabilities are expected, which generate decay waves with frequencies of the incident pump wave $\omega$ plus and minus the harmonics of the lower hybrid frequency $\omega_{LH}$ and the lower hybrid (LH) wave itself. With the electron cyclotron emission (ECE) receiver a spectrum of the decay waves with maxima at $\omega \pm n \omega_{LH}$ was measured for a resonant magnetic field of 1.25 T and is shown in Fig. 4. Note, that the pump wave is suppressed and the ECE-spectrum is clearly nonthermal since the density was twice the 70 GHz cut-off density. The low frequency LH-wave itself could be detected with a broad band loop antenna. A high degree of correlation between the high frequency decay waves and LH-wave was measured.

EBW's experience a cut-off layer ($N \to 0$) at the upper hybrid resonance (UHR) surface (see Fig. 1), which in the nonresonant or higher harmonic ($n \geq 1$) field totally encloses the inner plasma. The radiation is then trapped inside the plasma like in a hohlraum. The EBW is either reflected at
the UHR in the case of an oblique angle of incidence or is back converted to the X-wave which is converted again to the EBW at its next contact with the UHR. The only way that radiation can escape out of the Plasma is the small angular window for O-X- and X-O-conversion, respectively. In the absence of an electron cyclotron resonance in the plasma the EBW's may be absorbed due to finite plasma conductivity after some reflections at the UHR-layer. Nonresonant heating was clearly observed at magnetic fields up to 2.0 T. At the maximum field the plasma energy content increased by about 1.5 kJ compared to a similar discharge with NBI only (see Fig. 5). Two 70 GHz beams in O-mode polarisation (110 kW power each) were launched with an angle of 40° with respect to the perpendicular launch into a NBI (800 kW) sustained target plasma with a central density of 1.6 \times 10^{20} \text{ m}^{-3} and a central temperature of 560 eV. More than 80% of the heating power was found in the plasma if the power scaling of the energy confinement (P<sup>0.6</sup>) was taken into account. Thus O-X-B-heating turned out to be very efficient.

Ray-tracing calculation were performed with newly developed code in order to get a more detailed insight into the O-X-B-scheme. Density, temperature and magnetic field profiles similar to that of a typical neutral beam sustained W7-AS plasma were used for model calculations for a straight plasma cylinder. We use the nonrelativistic hot dielectric tensor with a correction for electron ion collisions given by Stix [3] and an isotropic electron temperature. The ray trajectory in the x-z-plane is shown in Fig. 6. The beam is launched from the low field side (LFS) and propagates through the cut-off, where it is converted into X-mode. Then it moves back to the UHR-layer, where the X-B-conversion takes place. The EBW’s are absorbed near the cyclotron resonance at the plasma centre. A small fraction of the beam power is lost at the UHR due to finite plasma conductivity. The power deposition zone for resonant heating strongly depends on the magnetic field and the electron temperature, but central power deposition seems possible. In calculations for the nonresonant case after six passes through the plasma more than 40% of the beam power is absorbed due to finite plasma conductivity.

**In conclusion:** Efficient O-X-B heating with 70 GHz electron cyclotron waves was clearly demonstrated for the first time for resonant and nonresonant fields at W7-AS. Both, the angular dependence of the O-X-conversion and the parametric instability which is typical for X-B-conversion could be experimentally verified. Density fluctuations at the O-X-conversion layer play a significant role in the O-X-B-process and need to be taken into account.

With a newly developed three dimensional ray-tracing code for EBW's and improved measurement techniques of the power deposition profiles further investigations of the O-X-B-heating are envisaged to explore the potential of resonant and nonresonant O-X-B-heating for routine high density operation.

**References**

Figures:

Fig. 1: Refractive index $N_x$ versus $\omega^2/\omega_p^2$ for the O-X-B conversion process. The transition represents the connection of the X-mode and B-mode due to the hot dielectric tensor.

Fig. 2: Modified O-X-conversion in the presence of density fluctuations at the plasma cut-off layer versus normalised density scale length $k_D$ for different relative fluctuation amplitudes.

Fig. 3: Increase of the plasma energy content by O-X-B-heating versus the longitudinal vacuum refractive index $N_z$ of the incident O-wave.

Fig. 4: High frequency spectrum of the parametric decay waves generated in the O-X-B-process. The incident wave frequency is 70 GHz and the LH frequency is about 900 MHz.

Fig. 5: Energy content (diamagnetic signal) of a NBI-discharge with and without nonresonant O-X-B-heating.

Fig. 6: Calculated ray trajectory in the x-z-plane and relative beam power for resonant O-X-B-heating.
Neoclassical Transport Predictions for Stellarators in the Long-Mean-Free-Path Regime

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The off-diagonal term in the neoclassical transport matrix, which is related to the particle flux, becomes essential in the stellarator long-mean-free-path (LMFP) regime. Strong temperature gradients can drive the density profile hollow, i.e., a positive density gradient related to the diagonal term in the transport matrix has to compensate this off-diagonal drive in order to fulfill the particle balance. As a consequence, central heating with peaked temperature profiles can make an active density profile control by central particle refueling mandatory. This effect will become essential for the larger stellarator devices of the next generation since recycling as well as gas puffing can affect only the plasma edge region of typically a few centimeters. This neoclassically predicted outward particle flux driven by the temperature gradients is experimentally confirmed in W7-AS discharges in the LMFP regime [1].

Necessity of an Active Density Profile Control

With a particle source, $S_p$, within the bulk plasma, e.g., by NBI and/or by pellets, a particle flux density: $\Gamma_{ex} = \frac{1}{r} \int_0^r r S_p \, dr$, is externally driven. Then, the ambipolarity condition, $\Gamma_e = \Gamma_i = \Gamma_{ex}$, with the neoclassical particle fluxes is given by

$$\Gamma_{ex} = -n \{ D_{11}^{e,i} \left( \frac{n'}{n} \pm \frac{E_r}{T} \right) + D_{12}^{e,i} \frac{T'}{T} \}, \quad (1)$$

with $\pm$ (−) for electrons (ions). Here, $T_e = T_i = T$ and $n_e = n_i = n$ was assumed for simplicity. The neoclassical transport coefficients, $D_{jk}^{\alpha}$ (with $j, k = 1, 2$ and $\alpha = e, i$) are obtained by energy convolution of the mono-energetic transport coefficients. For various magnetic field configurations, the databases of these mono-energetic transport coefficients calculated by DKEs code [2] are fitted based on traditional analytic theory [3] with axisymmetric contributions in the plateau collisionality regime taken into account.

Inverting eq. (1) leads to:

$$\begin{bmatrix} n'/n \\ E_r/T \end{bmatrix} = -\frac{1}{2} \begin{bmatrix} D_{12}^{e,i} + D_{11}^{e,i} \frac{T'}{T} & \Gamma_{ex} \begin{bmatrix} 1 \\ 2n \end{bmatrix} \frac{1}{D_{11}^{e,i}} \pm \frac{1}{D_{11}^{i,e}} \end{bmatrix}, \quad (2)$$

where the ratio of the transport coefficients is much less sensitive to the radial electric field, $E_r$, than the $D_{jk}^{\alpha}$ itself. Several roots of eq. (2) with respect to $E_r$ may exist: the “electron root” at large $E_r > 0$ with both the electron and ion transport coefficients being significantly reduced, the “ion root” at moderate $E_r < 0$ (for $T_e \approx T_i$) where mainly the ion $D_{jk}^{i}$ are decreased, and an unstable root inbetween. The 1st term of the r.h.s. of eq. (2) (with the + sign for $E_r$) is typically positive, and the “electron root” is forced for very small $\Gamma_{ex}$. For large $\Gamma_{ex}$, the “ion root” is obtained due to $D_{11}^{i} < D_{11}^{e}$. Assuming “pure” collisionality regimes, the normalized off-diagonal terms, $\delta_{off}^{\alpha} = D_{12}^{\alpha}/D_{11}^{\alpha}$, are easily obtained: $\delta_{off}^{e} = 7/2$ for the $1/\nu$ regime, $\delta_{off}^{i} = 3/2$ in the plateau regime, $\delta_{off}^{e} = 1/2$ for the $\sqrt{\nu}$ regime, and finally, $\delta_{off}^{i} = -1/2$ for the tokamak-like $\nu$ regime. As these regimes overlap in the energy convolution, the values of $\delta_{off}^{\alpha}$ and $\delta_{off}^{\alpha}$
given in Fig. 1 for an ambipolar particle flux $\Gamma_{ex}$ estimated from the condition $n' = 0$ reflect mainly the dependence on $E_r$. With decreasing collisionality ($\propto n/T^2$) the electrons enter the deep $1/\nu$ regime since the effect of the “ion root” $E_r$ on the $D_{jk}^i$ is small. The ion coefficients are mainly determined by the $\sqrt{\nu}$ regime which is very pronounced for the high mirror advanced stellarator configuration under consideration [3].

For the typical LMFP conditions, $\delta_{off}^e > \delta_{off}^i$ holds with the tendency of driving the density profile hollow. For the case of no central particle refuellng, i.e., $\Gamma_{ex} = 0$, eq. (2) gives for the pressure gradient

$$p' = -\frac{1}{2} (\delta_{off}^e + \delta_{off}^i - 2) n T',$$

with $p' > 0$ (for $T' < 0$) if $\delta_{off}^e + \delta_{off}^i > 2$ which holds in the stellarator LMFP regime. (Please note in this context, that $\delta_{off}^e + \delta_{off}^i \approx 0$ (or even negative) in the deep tokamak banana regime.) An inverted pressure profile, i.e., $p' > 0$, is in strong conflict with the MHD stability condition based on magnetic well, $V'' < 0$. Then, the typically stabilizing term, $p' V''$, becomes destabilizing. Consequently, the condition $p' < 0$ is mandatory leading to the requirement of an active density profile control, i.e., sufficiently large $\Gamma_{ex}$.

**Quantitative Estimates**

An estimate of the necessary particle refuellng rate can be obtained in a local (i.e., at a fixed radius, $r$) solution of the ambipolarity condition for a required $n'/n$, for given density and “heating power”. Here, the “heating power”, or more precisely, the total heat flux density, $q_t = q_e + q_i$, over the flux surface of effective radius, $r$, given by

$$q_{e,i} = -n T \left\{ D_{e,i}^{2} \left( \frac{n'}{n} \mp \frac{E_r}{T} \right) + D_{e,i}^{2} \frac{T'}{T} \right\},$$

is used. For the example shown in Fig. 1, a W7-X configuration with high toroidal mirror ($\approx 10\%$) [4] was selected. At about half the plasma radius ($r = 0.27$ m, $R = 5.5$ m), a (normalized) temperature gradient $r T'/T = 1$ was assumed which corresponds to a fairly peaked $T$ profile. The physical dependence of all the results on the local $T'/T$, however, turned out to be fairly small (e.g., a steeper $T'/T$ decreases $T$ at given heat flux $Q_t$). Full density control with $n' = 0$ was assumed. The necessary refuellng rate, i.e., the total ambipolar particle flux, scales roughly with the heating power for the 3 densities in Fig. 1. Please note in this context, that full NBI heating with a mean energy of 60 keV just fulfils this request ($\approx 10^{20}$ particles/s and per MW heating power). The temperature increases only slightly with $Q_t$ in the LMFP regime, nearly independent on density. At the high $Q_t$ values, the electron transport coefficients deeply within the $1/\nu$ regime exceed the ion ones, and, consequently, the $E_r$ of the “ion root” ($E_r < 0$) decreases since only a small $E_r$ is sufficient to reduce the $D_{jk}^i$ to the electron level ($D_{e2}^i \approx D_{22}^i$ at the low collisionalities, $\nu^*$). Consequently, the density dependence of the heat flux disappears in the deep $1/\nu$ regime. For both the normalized quantities, $\delta_{off}^e$ and $\delta_{off}^i$, as well as for the “convective” term, $T \Gamma_{ex}/q_t$, versus the collisionality $\nu^* \propto n/T^2$, an additional density dependence nearly completely disappears with the ambipolar $E_r$ taken into account.

**“Electron Root”**

Only for the lower densities, an “electron root” was found from the ambipolarity condition. In order to decide if this root can be realized, additional thermodynamic arguments have to be considered. On the basis of the radial force balance with a shear viscosity term included, a generalized heat production which has to be minimized is derived [5].
Fig. 1: Ambipolar particle flux (n’ = 0 assumed), temperature (T_e = T_i), ambipolar E_r vs. heat flux, Q_l (upper plots, from left to right), the norm. off-diagonal transport matrix terms, $\delta^{\alpha}_{\text{off}}$ and $\delta^{\alpha}_{\text{off}}$, the heat diffusivities, $D^{\alpha}_{22}$ and $D^{\alpha}_{32}$, and (for reference) the norm. “convective” heat flux $\Gamma^{\text{ex}}_e/Q_l$ vs. collisionality, $\nu^\prime$ (lower plots, from left to right) for the high mirror W7-X configuration at an effective radius of $r = 27$ cm. Densities ($n_e = n_i$): $5 \times 10^{19} \text{ m}^{-3}$ (•), $1 \times 10^{20} \text{ m}^{-3}$ (●), and $2 \times 10^{20} \text{ m}^{-3}$ (★).

Euler-Lagrange form of this variational principle leads to a diffusion equation for the radial electric field which is suited for integration in a predictive neoclassical code. For a “local” analysis, i.e., at one radius, however, the minimization of the generalized heat production with respect to the shear layer position (assumed to be sufficiently narrow) leads to the condition

$$\int_{E_r}^{E_r^*} (\Gamma_i - \Gamma_e) dE_r = 0,$$

(5)

with $E_r^*$ ($E_r^\prime$) being the “ion” (“electron”) roots. On the other hand, with the integral of eq. (5) being positive, the “ion root” will be realized. From this argument, the “electron root” solutions of Fig. 1 cannot be expected. In the integral of eq. (5), however, the assumptions for the “local diffusive ansatz” in the neoclassical theory are violated at least for the ions at $E_r \approx 0$, and direct losses have to be taken into consideration. From this point of view, the usual neoclassical ansatz may lead to an underestimation of $\Gamma_i$ for very small $E_r$ supporting the prediction of the “ion root”. Furthermore, a Kelvin-Helmholtz-like instability may be driven at the highly localized poloidal shear layer and may suppress the “electron root” feature. As a consequence, neoclassical transport predictions should not rely only on the optimistic prospects of the “electron root”.

“Ion Root”

For the “ion root” in the deep LMFP regime (with $D^{i}_{jk} \approx D^{\phi}_{jk}$), the total heat flux

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scales with \( T^{9/2} \), comp. Fig. 1. This unfavourable scaling makes stellarator optimization with respect to neoclassical confinement mandatory. For reactor scenarios based on the "ion root", very high densities at moderate temperatures are favourable. However, the negative \( E_r \) may lead to the problem of impurity accumulation. Assuming full density profile control with \( n' = 0 \), the "ion root" \( E_r \) can be estimated from \( \Gamma_i \approx 0 \) (since \( \Gamma_i(E_r) = \Gamma_i(E_r) \ll \Gamma_i(0) \)): \( E_r \approx \delta_{\text{off}}^i T' \). For the impurities, \( Z \), stationary conditions with \( \Gamma_Z = 0 \) (no inner sources) leads to the highly peaked profiles

\[
\frac{n_Z'}{n_Z} \approx (Z \delta_{\text{off}}^i - \delta_{\text{off}}^Z) \frac{T'}{T},
\]

since \( Z \delta_{\text{off}}^i - \delta_{\text{off}}^Z \gg 1 \) for high \( Z \). It seems to be unlikely that \( Z \delta_{\text{off}}^i - \delta_{\text{off}}^Z < 0 \) (which is the case in the deep banana regime in tokamaks) can be achieved by stellarator optimization.

The negative radial electric fields of the "ion root" result in a strong inward term for the high \( Z \) impurities. For the "electron root", on the contrary, no accumulation problems are expected.

Predictive Neoclassical Transport Codes

Neoclassical theory for fairly general stellarator configurations seems to be sufficiently developed, so that the predictive neoclassical transport modelling is the natural next step. A first attempt was done by implementing the Astra code [6] in a stellarator specific version [7]. The neoclassical transport matrix with the analytical representation [3] is used. The ambipolar \( E_r \) is obtained by direct iteration which is only stable for the "ion root". So far, the diffusion equation for \( E_r \) (corresponding to the radial force balance with the shear viscosity included) is not implemented. This problem is being treated by an other code which is still under development.

This stellarator specific Astra code version will be used to describe the transient phenomena in case of pellet injection used for the necessary active density profile control. In particular, the refuelling rate required to control the density profile in the bulk part of the plasma may be in conflict with the global density control if a transport barrier develops at the outer radii. Finally, on the basis of self-consistent density and temperature profiles together with the ambipolar radial electric field, the severe problem of impurity transport has to be treated.

Acknowledgements. We like to acknowledge helpful discussions with Dr. N.E. Karulin and Dr. J. Geiger.

References

Effect of the radial drift of trapped suprathermal electrons on the ECRH power deposition profile

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Introduction.
The ECRH power deposition in low density, high temperature plasmas has been analyzed at W7-AS for different heating scenarios (fundamental O-mode and second harmonic X-mode) and different magnetic configurations. These are characterized by a different size of the toroidal ripple on the magnetic axis in the toroidal position of power launching (see Fig. 1), allowing to study the influence of (toroidally) trapped particles. The analysis has been limited to the case of perpendicular on-axis heating.
The deposition profile has been estimated from the analysis of the electron heat transport in ECRH power modulation experiments, with the time dependent electron temperatures from ECE measurements. Peaked deposition profiles are usually obtained for both heating scenarios from a 3D Hamiltonian ray-tracing code based on the assumption of Maxwellian electron distribution function (single pass absorption). The heat transport analysis predicts the same peaked absorption profiles, but additionally a much broader contribution is present, whose width and relative integral contribution with respect to the “thermal” peaked part depends on the particular heating scenario and magnetic configuration.
The effect of the magnetic configuration on the electron distribution function in the different heating scenarios being considered, has been clearly demonstrated by means of a non-linear 2D bounce-averaged Fokker-Planck (FP) code, valid for the simplified magnetic field geometry close to the magnetic axis of W7-AS [1].
In a heuristic approach, the broadening of the thermal power deposition profile is expected to be related to the radial transport (determined by the $\nabla B$-drift) of the locally trapped suprathermal electrons generated by the ECRH. The particles drift vertically in the local magnetic ripple, becoming passing particles by pitch angle scattering, and therefore contributing to the energy flux in the outer plasma region without direct ECRH deposition, through thermalization on the flux surfaces.

Fig. 1. Magnetic field strength on-axis, normalized to the resonant field at the ECRH launching position, versus the toroidal angle within one field period for $t_0 \simeq 0.345$. The solid line refers to the “standard” configuration of W7-AS, the dashed and the dotted lines correspond to the “minimum B” and the “maximum B” launching scenarios, respectively.

Results of the power deposition analysis.
In Fig. 2a, the results of the power deposition analysis for scenarios at $B_0 = 2.5$ T in the “standard” configuration of W7-AS are summarized. The power deposition is clearly broadened for the lower densities both for fundamental O-mode (70 GHz) and for
Fig. 2. Power deposition profiles from ECRH power modulation experiments versus the effective radius \( r \). a: \( B_0 = 2.5 \) T. The solid and the dashed lines refer to the fundamental O-mode heating (70 GHz), for \( n_e = 2 \cdot 10^{19} \) m\(^{-3} \) and \( n_e = 10^{19} \) m\(^{-3} \), respectively. The dotted and the dash-dotted lines are the corresponding results for the second harmonic X-mode heating (140 GHz). b: \( B_0 = 1.25 \) T, and \( n_e = 2 \cdot 10^{19} \) m\(^{-3} \). The solid line refers to the “standard” configuration, the dashed and the dotted lines to the “minimum B” and the “maximum B” launching scenarios, respectively.

second harmonic X-mode (140 GHz) launching. These findings support the picture of significant heat transport by suprathermal ripple trapped electrons since both the level of the suprathermal tail decreases and the collisional detrapping increases with density. The formation of the suprathermal tail in the ripple trapped electron distribution is expected to be more pronounced for X-mode launching [2], leading to a stronger broadening of the “effective” power deposition profile. For the O-mode scenarios, however, the lower single pass absorption and the effect of wall reflections may lead also to an enhanced broadening of the power deposition profile.

The effect of direct heating of ripple trapped electrons becomes more clear from Fig. 2b, which shows the results of a scan of the toroidal ripple, at \( B_0 = 1.25 \) T with second harmonic X-mode launching. In the “standard” case with 70 GHz launching, the deposition is broadened compared to the 140 GHz discharges of Fig. 2a, and this effect may be attributed to the stronger \( \nabla B \)-drift of the trapped electrons. In the “maximum B” scenario, the deposition profile derived from the heat wave analysis is fairly close to the ray-tracing results, and the broadening of the deposition is of minor importance. For the “minimum B” scenario, a stronger broadening is found. All results on the power deposition are consistent with the heuristic model given above.

**Convective Fokker-Planck model.**

In order to simulate the broadening of the thermal power deposition profile, a simple FP model has been used, which describes the radial \( \nabla B \)-drift of the toroidally trapped suprathermal electrons generated by the ECRH. In a first approach, collisional or collisionless detrapping can be treated as a loss term, i.e., the detrapped electrons are assumed to thermalize on the flux surfaces. Taking further trapping of these detrapped particles into account leads to a kind of diffusive modelling (with inward and outward \( \nabla B \)-drifts). Neglecting this effect gives a convective model and the ECRH driven deviation of the trapped particle distribution from the Maxwellian defines the initial value for this convective FP.
model. Both an analytical model and the solution of a time-dependent bounce-averaged FP code \cite{1} are used for the initial distribution function. The drift of the suprathermal trapped electrons generated by the ECRH is described by means of the FP equation

$$< (v \nabla B)_z >_B = \frac{\partial f_e^l}{\partial z} = C_{\nu}(f_e^l) = C_{\nu}(f_e^l) + C_{\lambda}(f_e^l),$$

where \( f_e^l \) represents the bounce-averaged distribution function of the trapped particles, \( \nabla B \) the drift velocity due to the gradient of the magnetic field, and the angular brackets denote the bounce-averaging procedure. \(< ... >_B \equiv \int ... (ds/\tau_B) \int (ds/\tau_B) \), \( s \) being the coordinate along the magnetic field lines.

Stationary conditions are assumed. The presence of a radial electric field is omitted, so that the drift of the toroidally trapped electrons is mainly in the vertical direction \( z \). The model is valid in the bulk part of the plasma axis, where the radial electric field is negligible. The collision operator is linearized by assuming a Maxwellian background. This assumption is quite reasonable outside of the ECRH deposition zone. The linearized collision operator, \( C_{\nu}(f_e^l) \), is written as the sum of a diffusive term in \( w = ne^2/2T_e(z = 0) \) (\( \times \) energy), and a diffusive term in \( \lambda = v^2 A/B^2 B_0 \) (\( \times \) magnetic moment)

$$C_{\nu}(f_e^l) = \frac{2\nu(z)}{\sqrt{w}} \frac{\partial}{\partial w} \left\{ 2\eta(w_z) - \eta(w_z) \right\} f_e^l + \frac{T_e(z)}{T_e(0)} \frac{\partial}{\partial w} \eta(w_z) f_e^l,$$

where \( \nu(z) = \frac{\eta(w_z)}{w^3/2} \frac{\partial}{\partial \lambda} \left[ < \frac{1 - \lambda \beta}{\beta} >_B \lambda \frac{\partial f_e^l}{\partial \lambda} \right], \)

respectively, with \( \beta(s) = B(s)/B_0, \nu(z) = \pi \sqrt{2} e^4 n_e(z) \log \Lambda / m^1/2 T_e^3/2 \) is the (local) collision frequency, with \( \log \Lambda \) the Coulomb logarithm, \( w_z = ne^2/2T_e(z), \) and

$$\eta(w_z) = \text{erf}(\sqrt{w_z}) - \frac{2}{\sqrt{\pi}} \sqrt{w_z} \exp(-w_z), \ \ \ \ \eta(w_z) = \text{erf}(\sqrt{w_z}) - \frac{\eta(w_z)}{2w_z}.$$ 

Eq. (1) is solved with difference schemes in \( w \) and \( \lambda \) (implicit difference formalism in \( z \)). The boundary condition \( f_e^l(\lambda = \lambda_{lc}) = 0 \) is used, where the function \( \lambda_{lc}(z) = 1/\beta_M(z), \) represents the width of the trapped particle region, for different vertical positions \( z \), \( \beta_M \) being the maximum value of the normalized magnetic field strength \( \beta \). The derivative of the distribution function at the loss cone boundary, \( \partial f_e^l/\partial \lambda(\lambda = \lambda_{lc}) \), measures the collisional loss of the trapped electrons, while the collisionless detrapping is determined by the rate of changing of the function \( \lambda_{lc}(z). \)

This model is well suited to describe the effect on the “broadening” of the ECRH power deposition profile of the loss cone size (i.e., the impact of the different magnetic field configurations), the magnetic field strength \( (v \nabla B \propto 1/B) \), and the electron density, which enters via the initial value problem as well as via the loss rate due to collisional detrapping (the collisional slowing-down is of minor importance).

**Simulations of W7-AS scenarios.**

In the computations for W7-AS scenarios, the loss-cone width \( \lambda_{lc} \) has been assumed constant. Fig. 3a shows the power, normalized to the initial absorbed power, which is deposited at outer radii, in dependence of various parameters. The results are relevant to the “standard” configuration \( (\lambda_{lc} \approx 0.97) \). The behavior can be simply explained by the dependence of the ratio between the drift time, \( \tau_D \), and the collision time, \( \tau_c = 1/v_c \), with respect to the magnetic field strength and the plasma parameters

$$\frac{\tau_D}{\tau_c} \propto n_e B_0/T_e^{5/2} w^{5/2},$$

having neglected the variations with \( z \) of temperature and density, and the weak dependence of the Coulomb logarithm on the electron density. The broadening of the power
Fig. 3. Fraction of the absorbed power deposited at the outer radii. a: "Standard" configuration. The solid line is the result for $B_0 = 1.25 \, T$, and $n_e = 10^{19} \, m^{-3}$, the dashed line for $B_0 = 2.5 \, T$ and $n_e = 10^{19} \, m^{-3}$, and the dotted line for $B_0 = 2.5 \, T$ and $n_e = 2 \times 10^{19} \, m^{-3}$. b: $B_0 = 1.25 \, T$ and $n_e = 10^{19} \, m^{-3}$. The solid and the dashed lines are the results for the "standard" configuration and the "minimum B" scenario, respectively.

Deposition turns out to be larger in the case of low density and high temperature, and for decreasing magnetic field strength. Moreover, from Eq. (2) it results that the power deposition profile has a very strong dependence on the energy localization of the initial suprathermal trapped particles distribution function. The higher the velocity, where the EC resonance region is localized, the bigger is the effect of broadening of the power deposition profile.

In Fig. 3b the results for the power deposition in the case of the "standard" configuration ($\lambda_e \approx 0.97$) and the "minimum B" launching ($\lambda_e \approx 0.90$), are compared. Observe that for increasing loss cone width, the power is deposited on a broader radial range, and the particles could even be lost at the plasma boundary, before being thermalized. In the opposite limit of a narrow loss cone, the particles are detrapped by the pitch-angle scattering in a very short time, and in our model contribute to the power deposition only in a narrow region close to the axis. The theoretical predictions of this simplified model for the broadening of the power deposition profile, in particular its dependence on the plasma parameters and the magnetic field strength, are therefore consistent with the experimental findings.

In the case of decreasing $\lambda_e(z)$, as it is the case, e.g., for the L2-stellarator, the detrapping effect of the pitch-angle scattering is weakened, so that the vertically drifting electrons remain trapped for a longer time [3]. As a consequence, less power is deposited than in the case of constant loss cone width. Depending on the rate of decreasing of $\lambda_e$ and on their energy, a significant fraction of the electrons may be lost at the plasma boundary. In this situation, it becomes important to take into account the presence of a radial electric field. Qualitatively, it can be observed that the effect of the electric potential is to deviate the trajectories of the particles inside the plasma, therefore increasing their confinement. Most of the absorbed power is released in any case in the central region of the plasma, with the deposition profile width being determined essentially by the initial loss cone size, and the velocity distribution.

References.
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The Edge Turbulence in the W7-AS Stellarator:  
2d Characterization by Probe Measurements

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Introduction

In the edge region of the W7-AS stellarator spatially and temporally resolved measurements of turbulent electrostatic fluctuations are performed using Langmuir probe arrays. They are intended to explore the limits of our physical model based on former investigations of the scrape-off layer of ASDEX [1]. A very high correlation of the ion saturation current fluctuations $\parallel B$ was observed implying that the interaction between plasma and target plates must be taken into account. The sheath conditions are added to a 2d fluid model which includes magnetic curvature, $E \times B$, and diamagnetic drifts. In linear order its predictions agree with the experiment in a wide range of discharge parameters [1].

Another interesting subject is the investigation of the nonlinear dynamics of the saturated turbulent state. The latter is characterized by an unpredictable formation and decay of structures. Observing the evolution of structures should therefore offer insights into the nonlinear dynamics. Simple global properties, as average velocities, can be determined from the spatial-temporal correlation function. More detailed information, e.g. concerning the question whether typical structures exist, may be obtained by studying individual structures. Our starting point of a quantitative classification of structures is the decomposition of fluctuation data in a sum of simple “events” (pulse-shaped in space and time) with individually determined spatial and temporal position, velocity, size, lifetime, and amplitude. A search for statistically significant deviations from the spatially and temporally uniform distribution of the events follows. An analysis of that kind was applied to spatially 1d plus time dependent data taken from measurements with a poloidal probe array: In the environment of events with a high magnitude/size ratio the floating potential exhibits a dipole-like conditional average which is oriented such that in its center the $E \times B$ drift is directed radially outwards. Because the conditional average of the ion saturation current has a maximum there a very high local particle flux must be associated with these structures [2].

Of course, without a minimum knowledge about the radial behaviour it would remain uncertain to what extent the 1d poloidal properties describe the fluctuations. Especially the possibility of radially moving structures must be considered. New measurements at W7-AS with a right-angled probe array consisting of 20 tips in poloidal and 8 tips in radial direction are addressed to these questions.
Correlations

Basic space-time characteristics of the fluctuations are contained in the correlation function

\[ \rho(d, \tau) = \frac{\langle \tilde{s}(d, t) \tilde{s}(x + d, t + \tau) \rangle}{\sqrt{\langle \tilde{s}^2(x, t) \rangle \langle \tilde{s}^2(x + d, t + \tau) \rangle}} \]  \hspace{1cm} (1)

where \( x, x + d \) are the spatial positions of two probes and \( \tilde{s} \) denotes the fluctuating part of a random process \( s := \tilde{s} + \tilde{\tilde{s}} \) as given by the measured quantities (floating potential). For the present analysis time intervals with constant discharge conditions and negligible probe movement were chosen. Thus, \( \tilde{s} \) may be regarded as stationary. Estimates of (1) are based on temporal averaging.

Correlation \parallel B

To study the behavior of the fluctuations along the field lines simultaneous measurements of the floating potential with a radially movable Langmuir probe array of 19 tips extending in poloidal direction and a static Langmuir probe at a distant toroidal position were carried out. Because the poloidal positions of the probes are fixed the rotational transform of the magnetic field \( \iota \) had to be adjusted such that the field line bundle crossing the fixed probe passed the accessible poloidal-radial range of the movable probe array. (By numerical field line tracing it was estimated that this constraint requires a rotational transforms near \( \iota = 0.253 \) which is somewhat smaller than typical values for W7-AS.) The corresponding connection length between the 19 tips array and the distant probe is about 6 m. Other plasma conditions relevant to these measurements are: \( B = 1.25 \text{ T}, \text{gas} \text{H}_2, \text{line averaged central density} \ 1 \times 10^{19} \text{ m}^{-3}, \text{ECR heating}: 170 \text{ kW}. \)

Between the 19 tips array and the distant probe a correlation of 0.92 was found representing a lower limit for the actual maximum correlation parallel \( B \). It might be even higher, if the field line was not hit precisely. Within the temporal resolution (0.5 \( \mu \text{s} \)) the peak of the correlation function with the highest peak value has zero time shift. Apart from that the poloidal-temporal correlation function between the 19 tips array and the distant probe looks very similar to the correlation function obtained from the 19 tips array itself. Correlation lengths, correlation times and propagation velocities are the same. A comparison of such correlation lengths \( \parallel B \) to the correlations lengths \( \perp B \) in the order of 1 cm suggests an essentially 2-dimensional structure of the fluctuations. Obviously the physical process responsible for the fast balancing along the field lines has only very little tendency to spread \( \perp B \).

Correlation in the radial-poloidal plane

Measurements concerning the radial-poloidal structure of the fluctuations are performed with a right-angled probe array consisting of 20 tips in poloidal and 8 tips in radial direc-
Figure 1: Cuts through the spatially 2-dimensional correlation function of the floating potential for constant $\tau$.

tion. The distance between adjacent tips is 0.25 cm; the diameter of the tips is 0.09 cm. From such time series discrete estimates of the 2-point-correlation function $\rho(d_{ij}, \tau)$ can be calculated where $d_{ij} = x_j - x_i$ denotes the radial-poloidal distance vector between the $i$-th and $j$-th probe. Due to poloidal homogeneity $\rho(d_{ij}, \tau)$ is only a function of the poloidal probe distance whereas inhomogeneity in radial direction implies that $\rho(d_{ij}, \tau)$ depends not only on the radial distance but also on the radial position. For the right-angled array the $d_{ij}$ form a rectangular grid. Figure 1 shows cuts through the discrete 2-point-correlation function of the floating potential for constant values of $\tau$. Discharge parameters were: $B = 2.53$ T, gas: D$_2$, line averaged central density $1 \times 10^{19}$ m$^{-3}$, ECR heating: 300 kW. Smoothness was obtained by interpolation between the spatial grid points. The 2d spatial cut through the correlation function at $\tau = 0$ exhibits an oblique structure which may be attributed to an oblique orientation of the fluctuations. Movies, i.e. $\tau$-sequences, of such 2d spatial cuts indicate further that the fluctuations propagate as well in radial as in poloidal direction. In figure 1 a propagation to the left, i.e. radially
inwards, can be seen. These findings have important consequences for the interpretation of spatially 1d resolved measurements: Radial velocity components of oblique structures appear as poloidal motion, if only the poloidal direction is observed which explains the similarity between the radial-temporal and poloidal-temporal correlation functions. Velocity components perpendicular to the direction of observation lead to apparently smaller correlation times or lifetimes of individual structures.

It should be noted that figure 1 reflects the situation at one certain radial probe position a few centimeters outside the confinement region. Radial profiles of parameters associated with the poloidal correlation function show the well known velocity shear layer where the sign of the poloidal velocity changes and the correlation lengths and times exhibit minima. Accordingly at other radial positions different spatially 2d correlation functions are observed.

Conclusion

In the edge region of the W7-AS stellarator fluctuations of the floating potential show a very high correlation along the field lines over distances in the order of 10 m with zero time delay similar to former results for the ion saturation current in the scrape-off layer of ASDEX. This confirms the idea that in the shadow of a limiter the turbulence is essentially a 2-dimensional process under the far-reaching control of the sheath conditions. A poloidal and radial propagation of the fluctuations seen in the 2d spatial correlation function sheds light on the limits of spatially 1d resolved measurements.

References