W7-AS: One step of the Wendelstein stellarator line

I. INTRODUCTION

Stellarators are the oldest toroidal confinement concept, conceived by Spitzer in 1951 in Princeton.\(^1\) Stellarators share the relevant technology with tokamaks and they contribute, in a unique way, to the physics of toroidal confinement. The conceptual advantages of stellarators are their potential for steady-state operation without the need for current drive and the lack of current driven instabilities. Stellarators proper operate free of inductive current and their equilibrium is not established in a self-activative way. Net current-free stellarators are free of disruptions and of conventional and neoclassical tearing modes. The reason is that the confining magnetic field originates primarily from currents in external coils. Therefore, stellarators do not need an active control of the plasma position and cannot suffer from its possible failure. Figure 1 shows the coil system of a classical \(l=2\) stellarator (\(l\) denotes the poloidal symmetry) with a set of toroidal and helical coils. Wendelstein 7-A (W7-AS) was built after this concept.\(^2\)

The intrinsic properties of stellarators, which provide their advantages, also give rise to their drawbacks. As a consequence of net-current-free operation, stellarators have to be of three-dimensional (3D) geometry to provide a nested set of constant-pressure surfaces and thus plasma equilibrium and confinement. The stellarator geometry and field depends on all three coordinates; this is an important difference to the axisymmetric tokamak, where properties are conceptionally independent of the toroidal angular coordinate. The consequences of the 3D stellarator field are direct losses of energetic particles and large deviations of thermal particle orbits from flux surfaces leading to excessive neoclassical fluxes under reactor conditions. The equilibrium \(\beta\) of a classical

\(\text{Received 6 October 2004; accepted 13 January 2005; published online 20 June 2005}

This paper is a summary of some of the major results from the Wendelstein 7-AS stellarator (W7-AS). W7-AS [G. Grieger et al., Phys. Fluids B 4, 2081 (1992)] has demonstrated the feasibility of modular coils and has pioneered the island divertor and the modeling of its three-dimensional characteristics with the EMC/EIRENE code [Y. Feng, F. Sardei et al., Plasma Phys. Controlled Fusion 44, 611 (2002)]. It has extended the operational range to high density (4 \(\times\) \(10^{20}\) m\(^{-3}\) at 2.5 T) and high \(\langle\beta\rangle\) (3.4\% at 0.9 T); it has demonstrated successfully the application of electron cyclotron resonance heating (ECRH) beyond cutoff via electron Bernstein wave heating, and it has utilized the toroidal variation of the magnetic field strength for ion cyclotron resonance frequency beach-wave heating. In preparation of W7-X [J. Nührenberg et al., Trans. Fusion Technol. 27, 71 (1995)], aspects of the optimization concept of the magnetic design have been successfully tested. W7-AS has accessed the H-mode, the first time in a “non-tokamak” and has extended H-mode operation toward high density by the discovery of the high-density H-mode (HDH), characterized by H-mode energy and L-mode-level impurity confinement. In the HDH-mode quasisteady state operation is possible close to operational limits without noticeable degradation in the plasma properties. High-\(\beta\) phases up to \(t_{\text{pulse}}/\tau_E=65\) have been achieved, which can already be taken as an indication of the intrinsic stellarator capability of steady-state operation. Confinement issues will be discussed with emphasis on the similarities to tokamak confinement but also with respect to distinct differences in impurity confinement. In the HDH-mode quasisteady state operation is possible close to operational limits without noticeable degradation in the plasma properties.

\(^{a}\)This paper was presented at the APS DPP conference 2002 in Orlando as a review paper.
\(^{b}\)Electronic mail: fritz.wagner@ipp.mpg.de
\(^{c}\)Riso National Laboratory, Riso, Denmark.
\(^{d}\)Institut für Plasma Forschung, University Stuttgart, Germany.
\(^{e}\)IOFFE-Institute, St. Petersburg, Russia.
\(^{f}\)Princeton Plasma Physics Laboratory, Princeton, NJ.
\(^{g}\)KFKI-RMKI, Budapest, Hungary.
stellarator is low owing to its rather large aspect ratio. A technical disadvantage of a classical stellarator reactor would be the set of large helical coils. This technical problem has been removed by the development of stellarators with modular coils like W7-AS (see Fig. 2).  

The three dimensionality of the stellarator field offers—together with the use of modular field coils—the possibility to produce specific 3D magnetic field configurations where the third (geometrical) degree of freedom is used for system optimization. Thus the optimized stellarator concept was developed to overcome the deficiencies of the classical stellarator.  

Second order pressure-driven currents [diamagnetic, Pfirsch–Schlüter (PS), bootstrap current] appear also in stellarators. Whereas the diamagnetic current meets the equilibrium condition, the force-free PS current, which affects the shape of the constant-pressure surfaces, can be minimized by stellarator optimization. This is one of the design criteria of W7-AS. For a specific class of optimized stellarators, also the bootstrap current can be nullified. This is one of the design criteria of the fully optimized Wendelstein 7-X stellarator.

The Wendelstein line of Institut für Plasmaphysik, IPP, was founded by von Gierke in the early 1960s and developed over a sequence of targeted devices by Grieger. The first device, W1-A (Ref. 6) ($R=35$ cm, $a=2$ cm), a racetrack like

FIG. 1. Plotted is a classical $l=2$ stellarator (W7-A) with toroidal coils and four helical windings. A flux surface is shown with a magnetic field line.

the C-stellarator, 7 did not show good confinement; this was attributed to its poloidal $l=3$ symmetry, which provides only low rotational transform in the plasma core.

W1-B (Ref. 8) demonstrated the superiority of the low-shear $l=2$ stellarator and indicated that the deviations from toroidal symmetry—the straight sectors of the racetrack—are unfavorable. The future Wendelstein devices were toroidally periodic with $n=5$ symmetry ($n$ is the toroidal periodic number). W2-A (Ref. 9) demonstrated the viability of this basic concept in a somewhat larger device ($R=50$ cm, $a=5$ cm).

W2-B (Ref. 10) was a stellarator with ohmic heating of hydrogen plasmas and it showed, in a hybrid mode of operation, that the plasma parameters of stellarators were equivalent to those of tokamaks of similar size. W7-A ($R=2$ m, $a_{eff}=9$ cm; see Fig. 1) was the first large stellarator, which contributed fusion relevant data. It demonstrated for the first time in a fusion device with relevant plasma parameters stellarator operation proper without induced current.  

The parameters obtained under net-current-free operation matched those of similarly sized tokamaks. The Wendelstein line continued with W7-AS (1988–2002), whose results will be presented here, followed by W7-X which will start operation in 2010.

II. THE W7-AS DEVICE

The major goals of W7-AS, the device which followed W7-A, were

(a) to test the modular coil concept;
(b) to demonstrate the effectiveness of the first steps toward an optimized stellarator design; and
(c) to develop an exhaust concept based on the natural island chain, which resides at the plasma edge for higher iota values and whose separatrix forms the plasma boundary.

W7-AS has additionally contributed to many other areas of stellarator and fusion research.

W7-AS was a flexible system comprising 45 modular coils, which produced both toroidal and poloidal field components with dominant $l=2$ and $l=3$ poloidal and a $n=5$ (pentagon-like) toroidal symmetry.  

In a toroidal period, the plasma shape varied from an ellipse ($l=2$) to a triangular shape ($l=3$) and back to an ellipse. The coil concept of W7-AS is shown in Fig. 2. W7-AS applied for the first time a modular coil system to confine a fusion plasma. An additional set of toroidal field coils enabled change of rotational transform; five separately operated modular coils at the corners of the pentagon (with standing ellipses as cross sections) allowed variation of the toroidal mirror ratio; the plasma position could be changed with vertical field coils; an ohmic system was used to modify rotational transform, to compensate the bootstrap current in order to maintain the preset edge rotational transform during the discharge (this was the general mode of operation) or to study hybrid plasmas and current driven instabilities. Five in-vessel control coil pairs, which allowed to change the $B_{5,9}$ field spectrum, were used to vary the edge island size (and along with it the connection length) employed for divertor operation.

FIG. 2. (Color). Shown is a stellarator with modular coils (W7-AS). The modular coils produce toroidal and poloidal field components. The additional set of toroidal coils is to experimentally change $\iota$; larger modular coils reside at the corners of the racetrack. As they are energized separately, they allow the variation of the mirror ratio along the toroidal direction. (OH-coil system, vertical field and divertor control coils of W7-AS are not shown.)
A complete review of the results obtained with W7-AS is under preparation, see also Ref. 11.

III. THE ISLAND DIVERTOR

W7-AS pioneered the island divertor for the low-shear Wendelstein line. Figure 3 shows a poloidal cross section with cuts through the upper and lower divertor modules. Five divertor module pairs (top–bottom) were installed at the five toroidal positions with elliptical plasma cross sections. The modules consisted of target and baffle plates which formed fairly closed divertor chambers. Diversion occurred with the edge islands, preferentially via the $n/m$ ($n=5; m=8, 9, 10$) island chains. The strike points were defined by the intersection of the outer island separatrix with the target plates. The divertor gave access to plasmas with high density at quasisteady-state conditions, good energy and low impurity confinement properties, large radiation from the plasma edge, and partial detachment. The scrape-off layer and the discrete divertor modules established a 3D edge geometry, which was modelled by the 3D edge transport code EMC3/EIRENE. 12

Island divertor configurations were studied at high density generally with neutral beam injection (NBI). The power fluxes across the separatrix were up to 0.25 MW/m² and matched those of prototypical poloidal field divertor tokamaks. The $P/R$ values (power/major radius), used as edge scaling parameter, 13 were about 1.5 MW/m. The divertor allowed the density to be increased from $2.5 \times 10^{20}$ m⁻³ (only transiently accessible with limiter) to $4 \times 10^{20}$ m⁻³, maintained in quasisteady-state form. At this density level, most of the recycling neutrals are ionized in the scrape-off layer except at detachment. Power is radiated via low ionization states of mostly carbon from the target plate material. However, strong flux enhancement under high-recycling conditions is not observed and also not expected. Unlike the poloidal field divertor, the main operational characteristics of the island divertor are determined by the strong perpendicular transport of parallel momentum. The flow to the target plates is impeded even at high scrape-off-layer (SOL) temperatures and without intense neutral gas interaction due to the proximity of counterflows on the different parts of the island-scarep-off layers, which annihilate momentum. Figure 4 shows the downstream density in the scrape-off layer in front of the target against the upstream density at the separatrix midplane. At low density, there is roughly a linear relation between divertor and plasma density; at a midplane plasma density of $5 \times 10^{19}$ m⁻³ and a similar divertor density rollover appears followed by detachment without an intermediate high-recycling regime. The EMC3/EIRENE code (solid squares) reproduces qualitatively the results and confirms the lack of a high-recycling regime and other observations or conditions adopted in the code like the mechanism of flow damping without atomic interference, the importance of cross-field transport in the SOL, and—at the onset of detachment—the jump in carbon radiation and the propagation of the ionization front from the target to the separatrix. 12

In the comparison of experiment and model, one has to be aware that the model yields average values of the divertor density whereas the measurements are carried out locally.

In Fig. 5 the power deposition at the target plates under attached and detached conditions are compared. Detachment is partial and those sectors remain attached, which are magnetically connected to the midplane of the plasma at the outside (flux-surface compression leads to high local heat flux at the midplane outside whereas carbon radiation cools the plasma at the cold inside). Detachment is restricted in parameter space and it requires the island edge structure and the divertor: if the separatrix–target plate distance is reduced to less than ~2 cm or the connection length exceeds ~150 m, stable partial detachment is replaced by impurity induced plasma collapse. The dependence of the stability of the detached plasma phase on geometrical parameters is explained by the EMC3/EIRENE code. 12,14 A detailed account of the divertor studies is given in Ref. 15.

IV. PLASMA HEATING

Plasma heating was done with electron cyclotron resonance heating (ECRH; 70 GHz, ≈0.8 MW in the early operation phase; 140 GHz, ≈2.4 MW in the later phase), NBI
heating [up to 2.8 MW absorbed power; in the last phase with all-co (or all-ctr) beam orientation], and ion cyclotron heating (ICRH; up to 1 MW antenna power). ECRH is the fundamental heating power of present and future Wendelstein stellarators. In the general operational model of W7-AS, where the bootstrap current was integrally compensated by ohmic current, the ohmic heating (OH) power could be neglected for all heating scenarios.

Table I shows the development of ECRH gyrotrons for the Wendelstein Stellarator line starting with 28 GHz on W7-A in the 1980s, on the basis of a Varian tube, with 70 GHz, again from Varian, with 140 GHz, 3 s from IAP/Gycom up to the 140 GHz, steady-state tube developed for W7-X in the collaboration of Thales/Forschungszentrum Karlsruhe (FZK)/École Polytechnique Fédérale de Lausanne (EPFL)/IPP, and by CIP. The development has demonstrated pulse lengths, limited by the power supply in the combination 0.9 MW for 3 min or 0.5 MW for 15 min, respectively. Spring 2005, the CPI gyrotron has delivered 0.85 MW for 30 minutes.

ECRH of W7-AS was operated under conventional ordinary- and extraordinary-mode conditions for plasma heating and to study current drive (though conceptionally not needed for stellarators, these studies are characterized by high accuracy because the driven current is directly measured free from a large OH background current). High-density operation necessitated the development of ECRF heating beyond the conventional cutoff conditions on the basis of the known OXB heating scheme. The high-density plasma core was ultimately heated by electrostatic Bernstein waves. Under optimum conditions a high heating efficiency (>80%) was achieved. The prerequisites for successful OXB heating at high efficiency were provided at W7-AS, viz., steep edge density gradients ($L_n = 0.5–1$ cm) and a low density fluctuation level at the edge ($\approx 10\%$) for effective O-to-X-mode coupling and a high plasma density ($n_e > 2.5 \times 10^{20} \text{ m}^{-3}$ for 140 GHz) for effective mode conversion. Bernstein wave heating could be demonstrated up to the fourth harmonic (140 GHz at 1.1 T). An example of OXB heating is shown in Fig. 6. This scenario was also used for current drive and heat-wave experiments. The inverse process was used for temperature profile measurements at ultrahigh densities via electron-Bernstein-wave emission. A detailed account of the ECRH studies on W7-AS is given in Ref. 19.

Different ICRH heating schemes with sufficiently high single pass absorption were successfully employed on W7-AS, such as second harmonic heating, minority heating, and mode conversion. Moreover, the toroidal variation of the magnetic field was used for beam-wave heating. Figure 7 shows an example of ICRF beach-wave heating at an antenna power of 0.5 MW. The heating efficiency is $\approx 80\%$.

<table>
<thead>
<tr>
<th>Year</th>
<th>Frequency (GHz)</th>
<th>Power (MW)</th>
<th>Pulse length (s)</th>
<th>Supplier</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>1983</td>
<td>28</td>
<td>0.2</td>
<td>0.04</td>
<td>Varian</td>
<td>W7-A</td>
</tr>
<tr>
<td>1984</td>
<td>70</td>
<td>0.2</td>
<td>0.1</td>
<td>Varian</td>
<td>W7-A</td>
</tr>
<tr>
<td>1988</td>
<td>70</td>
<td>0.2</td>
<td>3</td>
<td>Varian</td>
<td>W7-AS</td>
</tr>
<tr>
<td>1993</td>
<td>140</td>
<td>0.45</td>
<td>1</td>
<td>Gycom</td>
<td>W7-AS</td>
</tr>
<tr>
<td>1997</td>
<td>140</td>
<td>0.4</td>
<td>3</td>
<td>CPI</td>
<td>W7-AS</td>
</tr>
<tr>
<td>2002</td>
<td>140</td>
<td>0.9 (0.5)</td>
<td>200 (1000)</td>
<td>Thales</td>
<td>W7-X</td>
</tr>
<tr>
<td>2004</td>
<td>140</td>
<td>0.5</td>
<td>600</td>
<td>CPI</td>
<td>W7-X</td>
</tr>
</tbody>
</table>
V. STELLARATOR SPECIFIC ISSUES ADDRESSED WITH W7-AS

A. Partial optimization

The geometry of W7-AS has been partially optimized. The parallel currents were reduced in comparison to an equivalent classical \( l=2 \) stellarator \((j_{l=2})/j_{l=2}^{\text{AS}} = 0.5\) by strong shaping (strong elongation in the corners of the pentagon). As a result, the equilibrium is improved for high \( \beta \) due to the reduced displacement of the magnetic axis (reduced Shafranov shift). The experimental analysis of the Shafranov shift as a function of \( \langle \beta \rangle \) (via SX tomography) has fully confirmed this design property. Figure 8 shows the variation of the magnetic axis \( R_{\text{axis}} \) with \( \langle \beta \rangle \) up to 3.4\% and compares the experimental data with the result from free-boundary equilibrium calculations (NEMEC, solid line). Good agreement is observed. The inset compares the relative Shafranov shift of a classical \( l=2 \) stellarator with that of W7-AS and demonstrates the improved equilibrium.

The bootstrap current in W7-AS is tokamak-like: it causes rotational transform to increase. Figure 9(a) shows as an example the bootstrap current as it develops during the discharge. In Fig. 9(b) the measured bootstrap current \( I_{\text{b1}} \) is compared with the neoclassical prediction using the DKES code. The agreement is better than a factor 2. The scatter of the data is attributed to \( Z_{\text{eff}} \), which was not measured for most of the cases.

An important design criterion of W7-X is that the net bootstrap current is close to zero because the toroidal field Fourier component is related to the helical one (the bootstrap current related to helical curvature reduces rotational transform). Of larger relevance for W7-X than the W7-AS findings are the results obtained from the Advanced Toroidal Facility (ATF).\(^{27}\) The proper selection of dipole and quadrupole field components of ATF allowed the bootstrap current to go through zero and change sign in agreement with theory.\(^{28}\) This observation gives confidence that the bootstrap current of W7-X will be small in agreement with the optimization so that the equilibrium properties will not change much from low to high beta.

The impact of partial optimization on neoclassical effects is more difficult to demonstrate experimentally than the one on equilibrium. The neoclassical heat transport, which governs the plasma core in the long-mean-free path (LMFP) regime of W7-AS, is strongly affected by the ambipolar radial electric field, which may mask the potential role of the partial orbit optimization. The radial electric field was measured for many experimental conditions in W7-AS with different levels of turbulence. In the plasma core region it is well represented by the balance of the neoclassical fluxes and the experimental results could well be reproduced by the neoclassical DKES transport code.\(^{26}\) This applies, however, also to radial ranges, where turbulent fluxes dominate the neoclassical ones. The indirect conclusion is, as in many other cases also, that the turbulent fluxes are mainly of electrostatic nature and do not contribute to plasma ambipolarity.

FIG. 7. Diamagnetic energy of an ECRH/ICRH heated plasma. An ICRH heating pulse in the beach-wave mode is applied from 0.3–0.5 s. After 0.6 s, ECRH is switched off and the plasma is maintained by beach-wave ICRF heating alone.

FIG. 8. Shift of the magnetic axis with \( \langle \beta \rangle \) (experimental data and NEMEC calculation); inset: relative Shafranov shift in W7-AS compared to a classical \( l=2 \) stellarator.

FIG. 9. (a) Bootstrap current as it develops during a W7-AS discharge. (b) The bootstrap current as measured is compared with the calculated values employing DKES. Results from two field values, 1.25 T and 2.5 T, are shown.
For medium to high density, when $T_e = T_i$, the plasma neoclassical transport follows the ion root. The ion root is always developed at the plasma periphery but generally also governs the plasma core region. Striking neoclassical transport effects can be observed at low density with $T_e > T_i$ when the electron root develops. The development of the electron root (in case of W7-AS driven by nonthermal electron fluxes originating from ECRF heating) with strongly positive electric field gives rise to low core transport and thus to high electron temperatures displaying the profile characteristics of internal transport barriers. A case is shown in Fig. 10. In the radial range $r < 5$ cm, the ambipolar field was measured to be strongly positive ($\pm 400$ V/cm); beyond this radius, the ion root is established.\(^{29}\) The strong positive electric field reduces the neoclassical electron heat diffusivity from 10–20 m\(^2\)/s to 2–5 m\(^2\)/s. The $T_e$-gradients in the inner zone are remarkable: $\nabla T_e = 1$ keV/cm, $R/L_{Te}$ (major radius to $T_e$-gradient length) $= 60$; in the outer by periphery $R/L_{Te} = 40$. The reduction of heat diffusivity is mostly attributed to the electric field. Whether also the field gradient—e.g., the one in the transition zone from ion root in the edge to electron root in the core—contributed to this reduction, e.g., via decorrelation of turbulence and reduction of anomalous contributions could not completely be ruled out; on the other hand, it had also not to be evoked to explain the observations.

### B. Fast particle confinement

In stellarators, the 3D magnetic configuration can lead to particles on unconfined trajectories. The losses are specifically severe for energetic particles. In the optimization strategy, a remedy is to locally increase $|B|$ in zones of large curvature. Also in the design of W7-X, this effect is embedded (besides the improved fast particle confinement via the magnetic well at higher $\beta$). With the use of a fast ion loss detector, the confinement of fast ions, injected by a radial neutral injector, was investigated at W7-AS for different toroidal mirror ratios employing the separately energized corner coils (see Fig. 2), which allowed to vary the field ratio at $36^\circ$ (elliptical plane, strong curvature) to $0^\circ$ (triangular plane, straight sector).\(^{30}\) Figure 11 shows the loss current as a function of the mirror ratio. Low field in the region of large curvature leads to enhanced ion losses; high fields in this region give rise to low fluxes. The mechanism to expel trapped particles from the regions of larger field curvature, which would locally enhance drifts, seems to be an appropriate measure to improve the confinement of energetic particles in 3D-toroidal confinement geometry.

### C. Low global shear

The field system of Wendelstein stellarators is designed for low shear, $S$. The basic design idea is to avoid low-order rationalities inside the plasma and to utilize the empirically observed good confinement in the proximity of low-order rationals. This concept-specific feature of confinement is consistently observed in the Wendelstein stellarator line (see Fig. 12) for low and medium $\beta$ discharges. In the neighborhood of low-order rationals, particle and energy confinement can be good; in between they can be degraded.

In tokamaks, rotational transform decreases to the edge and the conventional tokamak shows large shear. Advanced tokamak scenarios rely on low-shear operation made possible by larger bootstrap current contributions. In these operational scenarios, low-order resonances have to be avoided. On the other hand, transport barriers are observed in close proximity to low-order rationals. In a tokamak, zones with good confinement are indicated by local regions with steep gradients; in low shear stellarators, the global confinement turns out to show a strongly modulated dependence on iota. The relation of low-order rational surfaces and ion or electron transport is a topic of ongoing debate (see Ref. 31 and references therein).

The benefit of a radial zone free of major resonances but with maximally possible shear has specifically been exploited in the W7-X design, where iota ($\iota = \iota/2\pi$) is selected around $\iota = 1$ and the maximal shear is chosen between $5/6 < \iota < 5/5$ or $5/5 < \iota < 5/4$. $\iota$ increases to the edge; this feature is favorable because reversed precession of trapped particles renders more stable conditions against the onset of turbulence. The low-beta tokamak has the opposite radial
variation of \( \iota \); tokamak operation at high \( \beta_{\text{pol}} \) with a strong bootstrap current leads, however, to the same favorable field characteristics in the plasma core with the additional benefit of neoclassical tearing mode suppression.

These zones of good confinement in the neighborhood of low order resonances are the ranges where generally the confinement studies are done in W7-AS. In these windows also transitions into improved regimes like the H-NBI regime (best developed close to \( \iota = 1/3 \)) or the quiescent H mode (around \( \iota = 1/2 \)) occur. The development of the H-mode displays a fine structure in \( \iota \) to be discussed in Sec. VII A. As consequence, there are three levels of confinement:

(a) the standard “normal” confinement in the neighborhood of \( \iota = 1/3 \) and \( 1/2 \) (these are the ranges, where the confinement database is established);
(b) the low, substandard level between the maxima;
(c) improved regimes with confinement beyond normal values (generally limited to specific \( \iota \)-windows).

At high \( \beta \), the detailed variation of the confinement with \( \iota \) disappears.

The confinement results of W7-AS of a \( \iota \)-scan at low density \( (n_e = 4 \times 10^{19} \text{ m}^{-3}) \) with predominant electron transport are shown in Fig. 13. The experimental results are compared with the scaling results from the ISS 95 (Ref. 32) and the W7-AS (Ref. 33) scalings. The scalings are only related to the data at the confinement maxima around \( \iota = 1/3 \) and \( 1/2 \). The details of the iota dependence are compared with a heuristic model for the electron energy transport.\(^{34}\) Three contributions are considered:

(a) a background of neoclassical transport;
(b) a term subsuming, e.g., electrostatic turbulence and, on top of it;
(c) a local contribution of enhanced turbulence, attributed to rationals with an amplitude, which depends on \( n \) and \( m \), the toroidal and poloidal mode numbers, and with a radial extent around the rationals, which depends on shear \( S \).

The model describes the experimental results quite well. Because of the complex interplay of confinement, pressure, and internal currents with \( \iota \), the behavior is highly nonlinear. This is seen experimentally but also borne out by the model. Bi-furcations can occur between different confinement levels. These were also successfully predicted by the model.\(^{34}\)

The spatially nonresolved level of small-scale turbulence (measured by microwave or \( \text{CO}_2 \) scattering) increases at the transition from “good” to “bad” \( \iota \)-windows. Only indirectly, from slow \( \iota \)-scans, the turbulence can be attributed to specific zones inside the plasma possibly associated with rationals.\(^{35}\) Localized magnetohydrodynamic (MHD) modes move radially toward the outside and slow down in frequency when \( \iota \) is increased; different frequency bands merge which correlates with the development of plateaus in the \( T_e \)-profile and a reduction in confinement; finally the fluctuations are detected at the edge by the local Li-beam diagnostics. As soon as the plateaus reach the plasma edge, the confinement has reached the low, sub-L-mode level.

D. Magnetic well

In a low-shear device it is the magnetic well, which is predicted to ensure MHD stability against pressure driven instabilities. The vacuum field geometry of W7-AS has a magnetic well in the range of 1–1.5%. In addition, a strong stabilizing effect results from the magnetic well induced by the diamagnetic current, which develops with increasing \( \beta \).
VI. TURBULENT TRANSPORT AND CONFINEMENT

The outer plasma zone, which determines the energy confinement time, is governed by electrostatic turbulence (in the “good” iota ranges). The confinement time observed in W7-AS is comparable to that of tokamaks and reproduces its main features like power degradation and bifurcations—specifically to the H-mode. W7-AS was the first “non-tokamak” to operate in the H-mode, which demonstrated the universality of this confinement mode. Figure 15 compares experimental $\tau_E$ values to those expected from different stellarator and tokamak scalings: the international stellarator scaling (ISS 95), the W7-AS scaling (AS), the Lackner–Gottardi scaling (L–G) and the tokamak ELMy H-mode scaling [IPB98(y,2)]. The data of Fig. 15 are taken from discharges at operational limits (highest electron and ion temperatures, density, $\beta$, confinement time, at detachment and for the longest steady-state discharge). To allow comparison with tokamak confinement, $\iota$, $B$, and the geometry are rephrased into an effective plasma current; the average elongation of W7-AS is 1.9. For data points below the line, the experimental confinement time is longer than expected from the respective scaling. Most scenarios have better than predicted confinement time apart from the high density and the detached cases. Owing to the high density, the effective plasma radius is reduced in these two cases (as noted from the $T_e$-profile), an effect which has not been corrected for. There is an obvious similarity in anomalous tokamak and stellarator confinement because the experimental data are equally well described by either tokamak or stellarator scaling. In both cases, global confinement is limited by gradient driven turbulence. The good agreement seems to justify the replacement of current $I_p$ by $B$ and geometrical parameters. The most important aspect of Fig. 15 is the demonstration of no striking degradation of confinement at operational limits (apart from the high density case because of a reduction of the effective plasma minor radius, see Sec. IX B).

Despite this global agreement, there are characteristic differences in tokamak and stellarator confinement.

A. Global $\tau_E$-scaling

The $\tau_E$ scaling of the W7-AS ECRH discharges at the L-mode level is: $\tau_E = B^{0.73} \rho^{-0.5} L^{0.52}$. The size scaling ($L$) is selected to have correct dimensions. It is close to the size scaling of ISS 95: $L^{2.86}$. Separate scans have shown that the scaling with the (effective) minor radius is $a_{eff}$ yielding a nearly linear scaling with major radius. Thus the confinement time approximately scales with volume. Also a distinct but not well resolved dependence on iota has been found in separate scaling experiments: $\tau_E \approx \iota^{0.4}$. This dependence has been confirmed by TJ-II and it is of high importance for W7-X, which is designed with $\iota$ close to 1.

The W7-AS scaling can be rephrased in dimensionless parameters: $B \tau_E \approx \rho^{-2.73} \beta^{-0.03}$. The scaling is between Bohm and gyro-Bohm scaling. The scaling of W7-AS data meets the Connor–Taylor constraint of low-$\beta$ plasma models. An analysis of the data set employing Bayesian probabil-
NBI heating has been applied to reversed shear conditions and confinement regimes. It is interesting to note, that for density depends on the working gas. See Sec. VII B for details of the transition from normal confinement to the HDH mode develops. Unlike tokamaks, this limit, the HDH mode develops, and the parameter $I_e/L_{Te}$ is shown in Fig. 16 (dots) to be strongly reduced and steady-state operation is recovered. The IPB98 model predicts that for ECRH leads to flat (or hollow) $T_e$ profiles inside the deposition radius. Figure 17 shows a case where the heating power location has been switched during the discharge from central to midradius and back to central heating. Off-axis heating occurs from 0.4 to 0.6 s.

D. Lack of $T_e$-profile resilience

A distinct difference to the electron transport as observed in tokamaks seems to be the orthodox response of the $T_e$ profile in W7-A and W7-AS to changes of the heating power and the power deposition. On W7-AS, off-axis heating with ECRH leads to flat (or hollow) $T_e$ profiles inside the deposition radius. Off-axis heating occurs from 0.4 to 0.6 s.

Figure 18(a) shows $T_e$ profiles in log-representation for ECRH core heated electron transport dominated discharges with heating power in a large power range varying from 0.25–2 MW at $n_e=4.8 \times 10^{19} \text{ m}^{-3}$. Figure 18(b) shows the $T_e$ gradient (midradius) and the parameter $R/L_{Te}$ for two densities. The $T_e$ gradient increases with heating power; $R/L_{Te}$ is larger than in tokamaks (but the aspect ratio of W7-AS is 11) and falls with increasing power because of a strong $T_e$ rise in the plasma periphery. For the 2 MW case, the power flux through the mid radius flux surface is about 0.25 MW/m².

Heat wave experiments by ECRH modulation yield generally $\chi_{pe} = \chi_{pe}$ for the electron heat diffusivities measured via heat pulse propagation or obtained from a conventional power balance analysis. Figure 19 shows the results from on- and off-axis power modulation. The phase of the $T_e$ wave drops symmetically to the inside and the outside of the off-axis deposition radius ($r/a=0.45$) [Fig. 19(a)]. The $T_e$ profile regime, the confinement is superior by a factor of 1.5 to 2.

B. Isotopic effect

No distinct confinement difference is observed between hydrogen and deuterium discharges (in all collisionality- and β-ranges, with $T_e/T_i$ large or close to 1). There is also no obvious isotopic dependence in the required conditions to transit into the H-mode. Therefore, the results given in Fig. 15 are obtained with H+ (and H0 injection). The IPB98(y, 2) scaling results in Fig. 15 are calculated with $A_y=2$!

The only obvious isotopic effect on confinement are the IPB98 results for this study did not yet contain the more recent high-β results.

C. Density scaling

Stellarators show a distinct density dependence in confinement time up to the limit of 1.5–2 x 10²⁰ m⁻³ (beyond this limit, the HDH mode develops). Unlike tokamaks, this general $n_e$ scaling does not seem to lose its importance in larger devices (LHD). Figure 16 summarizes the role of density on energy and impurity (aluminum tracer) confinement time. At lower densities the plasma is heated by ECRH (open circles), whereas at higher density (>8 x 10¹⁹ m⁻³) NBI heating has been applied (full squares). With NBI at a density beyond 1.5 x 10²⁰ m⁻³, the transition into the HDH regime occurs. The medium to low density cases are compared with the W7-AS and the ISS 95 scaling. In the HDH regime, the confinement is superior by a factor of 1.5 to 2. H⁺ results are shown for comparison (×). Also the impurity confinement time $\tau_I$ is shown in Fig. 16 (dots); it is measured by Al-laser blowoff. This parameter also scales strongly with density with the consequence, that beyond $n_e=5 \times 10^{19} \text{ m}^{-3}$ at $\tau_I/\tau_e \approx 20$ impurity accumulation occurs and no steady-state operation is possible. At the HDH transition, when $\tau_I/\tau_e \approx 70$, $\tau_I$ is strongly reduced and steady-state operation is recovered. The role of the stellarator line will critically depend on the impurity transport characteristics. W7-X will play a crucial role in elucidating this feature under more reactor relevant conditions. A detailed report on impurity transport in W7-AS is published in Ref. 44.

FIG. 16. (Color). Dependence of energy ($\tau_E$) and impurity ($\tau_I$) confinement times on the line averaged density ($\langle n_e \rangle=0.12 \text{ m}^{-3}$; $\tau_e=0.5$, $B_T=2.5 \text{ T}$, $P =0.65–0.75 \text{ MW}$). The transition into the HDH regime is indicated. Also the ratio of $\tau_I/\tau_E$ is given (dotted curve). The $\tau_E$ values with open symbols and the $H^+$ values are obtained at different $\langle n_e \rangle$ and scaled according to the $(a_0^2)$ to allow comparison.

FIG. 17. (Color). $T_e$ profiles of an ECRH heated discharge with on- and off-axis heating phases. Off-axis heating occurs from 0.45–0.65 s, the power deposition occurs at $r=0.12 \text{ m}$.
The experimental analysis shows that the thermally driven plasma core under central heating conditions without central particle source, because there is no Ware effect, is flat in the inside but steep in the outside [Fig. 19(b)]. In a mixed case with on- and off-axis heating (at the same total power of 1.2 MW), $\chi_{\text{HP}}$ and $\chi_{\text{PB}}$ could be analyzed [Fig. 19(c)]. Outside ECRH deposition both $\chi_e$ values are unchanged. For $r/a < 0.45$ only $\chi_{\text{HP}}$ could be estimated. It rises from $1 < \chi_{\text{HP}} < 2.5 \text{ m}^2/\text{s}$ with off-axis heating to $4 < \chi_{\text{HP}} < 12 \text{ m}^2/\text{s}$ for 0.6 MW central ECRH power. Also, these studies do not reveal an intrinsic transport mechanism, which is governed by turbulence onset at a critical gradient $L_{r,e}$. It is not clear whether the reversed shear profile and the different role trapped particles play is the reason.

But there are also similarities in transport between W7-AS and tokamaks, e.g., in the particle transport of the working gas which are not necessarily expected. One would expect that the density profiles of stellarators are flat in cases without central particle source, because there is no Ware pinch as stellarators are normally operated without external momentum input. Stellarator density profiles are flat in the plasma core under central heating conditions without central fueling. The gradients reside at the plasma edge in the range of the recycling source. With core fueling (NBI), density profiles peak as a consequence of the changed source distribution. With strong central ECRH heating, the flat density profile becomes even hollow. With off-axis ECRH heating and flat central temperature profiles, the core density profile however peaks without central source (see Fig. 20). Therefore, also in stellarators, there is a convective inward term.

These observations elucidate the effects which govern the particle transport: Particle diffusion, thermal diffusion, and a convective particle inward term play a role. Thermal diffusion causes flat density profiles at peaked $T_e$ profiles. The experimental analysis shows that the thermally driven ($I = \nabla T_e$) neoclassical flux component, which is outward directed in stellarators, can explain the hollow $n_e$ profiles of central ECRH heating. The convective inward term, caused by an $E_r/T_e$ term in the transport equation or resulting from the background turbulence, causes the density profile to peak when the central $T_e$ gradient is flat (owing to off-axis heating) and thermal diffusion does not enter. EIRENE code calculations show that the core fueling rate can indeed be ignored in these ECRF heated plasmas.47

VII. IMPROVED CONFINEMENT REGIMES

A. H-mode

In the experiment a fairly complex manifestation of the H-mode is observed, depending on the magnetic configuration, on density and heating power.

The H-mode of W7-AS shares with the one found in tokamaks the following characteristics:

(a) The H-mode develops most easily (low power) with separatrix edge.
(b) It can develop with grassy edge localized modes (ELMs), solitary large ELMs, with dithering and quiescent phases.
(c) The H-mode transition or back-transition and the appearance of ELMs are, like in tokamaks, marked by the $H_a$ traces and readily seen in all turbulence monitors.
(d) The edge fluctuation level strongly decreases at the transition.
(e) Steep parameter gradients develop at the plasma edge.
(f) The H-mode correlates with the development of an electric field well at the plasma edge. The transition occurs typically at a field of $E_r \approx 200$–300 V/cm and a field gradient $\nabla E_r \approx 100 \text{ V/cm}^2$; during the H-phase the well further decreases down to values of $E_r \approx 700 \text{ V/cm}$ and a field gradient $\nabla E_r \approx 200 \text{ V/cm}^2$.
The H-mode remains typically for a confinement time after the heating pulse has been terminated; then an L-mode back transition occurs.

ELMs originate from the plasma edge with a pivot point which resides about 1–3 cm inside the separatrix.

ELMs expel particles and energy and they last for about 200 μs.

The energy and particle content losses from larger, distinct ELM are typically 5%.

Quiescent H-phases are prone to impurity accumulation despite of broad density profiles.

The following differences to typical tokamak H-mode characteristics seem to be of relevance:

**a.** Judged by the presence of grassy ELMs, the H-mode can develop right at the beginning of the discharge without noticeable transition.

**b.** The transition into the quiescent H-mode occurs at a critical (power dependent) density and has an upper power limit. Because of the critical role of the density, a delay between the heating pulse and the H-transition cannot clearly be concluded.

**c.** The quiescent H-mode develops in small iota windows in the neighborhood of \( \nu \approx 0.5 \) when the plasma edge is bounded by an island separatrix.

**d.** At high density, the H-mode can develop in the form of the high-density H-mode (HDH) with good energy and—opposite to the conventional H-mode—low impurity confinement.

**e.** Around \( \nu \approx 0.34 \), under limiter conditions at low to moderate densities the so called H-NBI mode develops; the name has historical reasons because, originally, it seemed to be restricted exclusively to NBI heating. The steep gradient radius and the origin of small ELM-like fluctuations are located further in. Later, means were developed to initiate the H-NBI mode also around \( \nu \approx 0.5 \) (with the tendency to develop a regular ELMy H-mode) and with ECRH.

Figure 21 shows a typical NBI heated H-mode discharge. The transition occurs spontaneously at about 400 ms in a phase where the density has been increased up to a new plateau. In this phase of constant density, the energy content \( W \) keeps growing up to the transition when it further increases by about 50%. The H-mode transition is depicted by the \( H_a \) radiation; ELMs appear in the H-phase. At about 600 ms, NBI has been turned off. The plasma remains in the H-phase for about 20 ms; then an L-mode back transition occurs. This delay is a known feature of tokamak H-modes. The back transition causes an increase of \( H_a \) and a faster decay of \( W \) and the density. The tangent in Fig. 21 helps to guide the eye.

Figure 22 shows a quiescent H-transition. Plotted are \( H_a \), floating potential at two radii inside the separatrix, the edge ion temperature, and the perpendicular flow velocity (both measured by passive spectroscopy) and on the right side (for a similar discharge), \( \bar{B} \) and \( \bar{n} \) (from reflectometry). At the plasma edge the H-transition causes a rise in the ion temperature and the perpendicular flow; the electric field becomes more negative and the fluctuation level sharply decreases.

Figure 23 shows the H-mode variants with the 5/9 island separatrix as edge conditions in a \( P \) vs \( \langle n_e \rangle \) operational plane. H-modes have been obtained with ECRH and NBI down to the technical power limits of 0.25 MW. A possible power threshold is lower than in conventional tokamaks. The low value of W7-AS may reflect the low neoclassical damping of poloidal flow owing to the large aspect ratio. As spherical tokamaks seem to have a higher power threshold than conventional tokamaks, these differences of the three toroidal confinement concepts may point toward an intrinsic aspect ratio scaling of the H-mode power threshold. The power threshold scaling of Ref. 52, which includes the effect of aspect ratio as obtained from spherical tokamaks, yields 1 MW for W7-AS—at least a factor of 4 too high.
ELMy H-modes are obtained in a large parameter range. The quiescent H-mode is restricted in operational space with a tight coupling between density and heating power. The HDH mode (see Sec. VII B) can be accessed from ELMy phases or from the H*-mode. It is restricted to high density.

Figure 24 is another operational diagram spanning density and i. The data points show H- and L-mode plasmas and indicate three H-mode windows at specific i-values. Also shown is the density limit for the heating power of 0.4 MW (see Sec. IX B). H-modes can be operated close to the density limit which nominally varies with i because of the intrusion of edge islands, which modulate the plasma cross section and with it the heating power density, which is a scaling parameter of the density limit;53 see Sec. IX B). Indicated in Fig. 24 are also the ranges with large poloidal viscosity due to the specific field spectrum at the edge of W7-AS. Obviously, the quiescent H-mode cannot develop in regions with low order islands at the edge and, as a consequence, with strong poloidal damping.54 (A similar i-dependence of the H-mode transition conditions are observed in He-J,55 the H-mode windows do, however, not coincide with the conditions for low damping!)

Figure 25 shows for two separate discharges the variation of the spectroscopically measured radial electric field E_r, the diamagnetic term \( \nabla p_i/n_e \), deduced from the local parameters (\( T_i, n_e \), and \( n_C \), the measured carbon density), and the poloidal flow velocity \( v_\Phi \) resulting from the difference between field and diamagnetic component. \( v_\Phi \) is assumed to be zero at the plasma edge, which seems to be justified at the prevailing high edge densities; the few experimental checks on \( v_\Phi \) at the edge confirm this. For orientation, the \( H_a \) traces are also given. (Because of the uncertainty in the data, two cases at different parameters are shown to elucidate the common features.)
Both discharges develop a quiescent H-mode (No. 47108 at \(n_e=0.7 \times 10^{20} \text{m}^{-3}\) and \(P_{\text{NBI}}=0.4 \text{MW}\), No. 47121 at \(n_e=1.2 \times 10^{20} \text{m}^{-3}\) and 1.2 MW). The independently determined \(E_r\) and \(\nabla p_i/\epsilon n_e\) agree quite well up to the H-transition. This is also borne out in a larger database. The independently determined \(E_r\) and \(\nabla p_i/\epsilon n_e\) no longer match. Obviously, a \(v \times B\) term (we assume a \(v \phi \times B\) term and ignore a \(v \phi \times B\) term at the edge) develops which contributes to further reduce the (negative) electric field. It is possible that this term, which is given by the poloidal momentum balance, introduces the \(\epsilon\)-variation in the H-mode transition conditions. The agreement of \(E_r\) and \(\nabla p_i/\epsilon n_e\) in the L-phase and the disagreement in the H\(^+\) phase are shown in Fig. 26 in more detail.

Figure 27 expands the time scales for the transition and plots the L-H transition and the H-L back transition (after the NBI pulse has been switched off). The back transition is delayed and occurs about 12 ms after beam switchoff. Plotted are again the spectroscopically measured \(E_r\) supplemented by \(E_r\) deduced from Doppler reflectometry. The Doppler data require the presence of turbulence, as provided by the L-phase or during ELMs. Between ELMs, no signal is available. The poloidal propagation of turbulence is used to assess \(E_r\). The agreement of both methods is good; reflectometry provides the better time resolution. Plotted in Fig. 27 is also the power level of the turbulence (measured with the same probing \(\mu\)-wave beam at identical location). All data are taken in the vicinity of the profile pivot point from L- to H-mode.

The transition into the H-mode shows a preceding phase where \(E_r\) already deepens (the absolute value of the negative \(E_r\) is plotted) and the turbulence gradually reduces. The transition occurs typically at an \(\nabla E_r=10^3 \text{V/m}^2\). Interesting is the back transition. As soon as the beams are switched off at 0.6 s, \(|E_r|\) decreases and the turbulence level increases slightly, and scattering signal appears sufficient to measure the propagation of the fluctuations and to deduce \(E_r\). At the back transition, \(E_r\) and simultaneously the fluctuation power jump (note the logarithmic scale for the fluctuations). The jump in fluctuation power is consistently observed; the one in \(E_r\) does not seem to be a generic feature. Doppler reflectometry with temporal resolution optimized down to 10 \(\mu\)s shows that the fluctuation power and the fluctuation propagation velocity display anticorrelated oscillations in the vicinity of the transition, i.e., a decreased propagation velocity (respectively, a decrease in its shear) correlates with an intermittent increase of the fluctuation level.

In the second L-phase, the \(|E|\) field continues to decrease (the plasma energy content decays after termination of the heating power), the fluctuation level continuously increases until both saturate in the evanescent phase of the plasma. Both transitions exemplify the close correlation between \(E\) field or its gradient and turbulence.

The H-transition occurs or is lost under conditions, which are gradually approached in the pre- (respectively, the post-) phases.

The H-mode of W7-AS shows the features as typically observed in tokamaks: Generally transition during the heating phase favorably under separatrix conditions, dithers, delayed back transition, ELMs and quiescent phases, relation between \(E_r\) (or \(\nabla E_r\)) and the turbulence level, \(E_r\) is predominately but not exclusively determined by the diamagnetic contribution. Major differences seem to be the distinctly lower power threshold, a power dependent density threshold...
for the transition into the H-mode to become quiescent in NBI heated plasmas and—like in L- and H-mode confinement—no obvious isotopic scaling in the power threshold.

**B. High-density H mode**

At even higher density, the dynamics of the plasma changes once more. A transition into the HDH-mode occurs (Fig. 23). This mode is accessible from the quiescent H-mode (H+) or from NC (normal confinement, either L-mode or ELMy H-mode), which is, in this high-density regime, established by NBI with rather peaked density profiles. The transition into the HDH-mode can thus occur from a state with broad (H+) or a state with peaked density profiles. It always ends in a state with very broad density profiles. The HDH-mode is free of ELMs possibly the edge bootstrap current is small at the high edge collisionalities.

The transition from H+ to HDH is initiated by gas puffing to increase the density. The transition requires a critical density, which itself depends on heating power \( P_{\text{abs}} = 0.7 \text{ MW}; n_e = 1.5 \times 10^{20} \text{ m}^{-3} \); 1.4 MW: \( n_e = 1.8 \times 10^{20} \text{ m}^{-3} \); 2.45 MW: \( n_e = 2.2 \times 10^{20} \text{ m}^{-3} \). With the gas puff, \( E_r \) becomes smaller and, after a dwell time, HDH develops, the impurity radiation drops and moves to the plasma edge, and the power flow onto the target plates rises.

With beam fueling, the preceding phases are prone to impurity accumulation, increase of core radiation and discharge collapse irrespective of the density profile form (NC or H+). As soon as the HDH-mode is reached, the impurity confinement time rapidly drops (Fig. 16) and the plasma purifies itself. Figure 28 shows energy content, line density, and total radiation of different confinement regimes \( \text{H}^+, \text{NC} \) (ELMy H-mode in this case), HDH, detached plasma. \( \text{H}^+ \) and ELMy H-modes do not reach steady state; \( \text{H}^+ \) is subject to a rapid and the ELMy H-mode to a slow increase in radiation. HDH and detached plasmas reach steady state. The radiation profile strongly changes from core dominated to an edge dominated radial profile. The energy content of the ELMy H-mode has first a maximum because it is mostly ELM-free but decreases thereafter by a factor of two due to increasing ELM activity. Under detached conditions, the edge radiation is close to 100%; the energy content does not further rise with density because detachment leads to a reduction of the effective plasma cross section. In summary, at high density, small changes decide on the confinement regime which develops operationally.

The change in energy confinement time at the transition to HDH depends strongly on the pre-phase: Starting from NC, \( \tau_E \) rises at the transition (confinement enhancement factor \( f_H \approx 1.7–2 \)), starting from the H- mode, the confinement remains about the same (Fig. 16). The ratio of \( \tau_I/\tau_E \) drops from 50 to 2, demonstrating the striking impact of the HDH conditions on the impurity confinement. The HDH \( \tau_I/\tau_E \) ratio would excellently meet ignition conditions.

It is difficult to understand the sudden decrease in impurity confinement and simultaneous increase of \( \tau_E \) going from NC to HDH. At the transition, \( \tau_e \) and \( \tau_I \) are no longer conforming to each other: \( \tau_e \) increases, \( \tau_I \) decreases. A canonical relation observed basically in all confinement transitions (with and without edge or internal transport barrier, from broad to peaked density profiles) and reflected by major \( \tau_E \) scaling parameters (power, ion mass in case of tokamaks) seems to be violated. Obviously a specific but (unfortunately) not yet unravelled mechanism sets in at the edge, which discriminates between energy and impurity transport.

Both for tokamaks and stellarators the proton density profile form plays an important role for the ratio of \( v_{\text{in,imp}}/D_{\text{imp}} \) which governs impurity transport and steady-state characteristics. Qualitatively, it may be understandable that the broadening of the density profile at the transition from NC with peaked \( n_e \) profiles to HDH with broad ones removes the impurities. Indeed, laser-blowoff (LBO) measurements show that \( v_{\text{in,imp}}(a) \) is the quantity; which changes strongly at the transition to HDH. Applying a simplified transport model \( [D_{\text{imp}}(r) = \text{const}, v_{\text{in,imp}}(r) = v_{\text{in,imp}}(a)(r/a_{\text{eff}})] \), \( v_{\text{in,imp}}(a) \) was found to be strongly reduced from \(-10 \text{ m/s} \) to \(-2.7 \text{ m/s} \) from NC to HDH whereas \( D_{\text{imp}} \) stays rather invariant at 0.1 m²/s. The question remains, which mechanism causes \( v_{\text{in,imp}}(a) \) at the edge to strongly change whereas \( D \) and \( \tau_E \) remain rather invariant.

The HDH regime is close to the operational density limit of W7-AS and if the absolute value of \( n_e \) is governing the transition parameter, it may only be accessible by high-field tokamaks. For normal tokamaks, the above density values are close to or beyond the Greenwald limit. But the HDH-mode may be related to the EDA-mode of Alcator C-mod operating at high field and high density, which appears with rather similar features (H-mode energy, L-mode impurity confinement, no ELMs).

During the EDA-mode of C-mod, a quasicoherent instability appears at the plasma edge, which is interpreted as resistive ballooning mode connected to the magnetic particularies of the X-point. There is no evidence of a similar instability at the edge of HDH plasmas in W7-AS. This mode has been searched for with all available edge diagnostics. Its presence would have been noted. [This mode has also been observed at the edge of ELM-free H-modes of the (old) ASDEX at lower densities; in ASDEX, the quasicoherent mode could not prevent impurity accumulation in quiescent phases.]

The HDH regime allows steady plasma operation and it
is rather robust. This mode can easily be established for all powers above 0.7 MW (power flux across the separatrix: 0.06 MW/m²) and it is established for the highest $\beta$-values achieved in W7-AS. Considering the advantages of high densities for divertor operation, the HDH-mode may be the preferred operational mode for W7-X (depending on the necessary transition power and the accessibility possibly restricted by the density limit). Then the role of the collisionality $\nu$ in the transition conditions and further characteristics of the HDH-mode can be explored. In case a more ubiquitous character of the HDH mode emerges in the W7-X operation, it might be the favored operational mode in the future because it combines good energy confinement and low impurity confinement without the need of ELMs. The importance to reach the HDH regime is obvious from the fact that W7-AS did show impurity accumulation beyond a density of $5 \times 10^{19}$ m⁻³; only the HDH regime at high density provided steady-state conditions.

Assuming that the transition density into the H-mode scales with power density $P/\text{Vol}$ like the density limit does (see Sec. IX B), the following scaling is obtained from the—however scarce—W7-AS data: $n_{\text{crit, HDH,20}} = 1.66 (P_{\text{abs,Vol}})^{0.3}$. Applying the HDH density scaling and the density limit scaling as found on W7-AS, $n_{\text{crit, HDH}} \leq n_{\text{DL}}$ for $P_{\text{abs}} > 3$ MW; the accessibility of the HDH regime is not restricted by the density limit. In case a minimum density of $1.5 \times 10^{20}$ m⁻³ is necessary in W7-X for the HDH regime like in W7-AS, a heating power of 30 MW is necessary, a level which is foreseen in stage-II heating.

C. H-NBI mode

At $\nu = 0.34$, the other $\nu$-window with good confinement, the conventional H-mode with steep gradients right at the separatrix does not appear probably because only limiter operation is possible. In this $\nu$-range another good confinement mode, termed the H-NBI mode, develops at low to medium densities, again outside the operational windows for the conventional quiescent H-mode. Because of the large plasma cross section ($a_{\text{eff}} \approx 0.18$ m), the best confinement times have been obtained in this regime. This mode does not develop in a fast transition but develops gradually (typically within 100 ms) in a time longer than the confinement time. The transition occurs at constant density and is characterized by a self-sustained evolution of the plasma temperatures, which increase, and temperature profiles, which expand.

The characteristic element of the H-NBI mode is a narrow density profile with gradients shifted inward and low $n_e$ at the edge. These conditions allow for broad profiles of $T_e$ and $T_i$ with steep gradients in the region of lower densities, i.e., outside $r/a \approx 0.7$ where approximately $T_i = T_e$. Both high $T_i$ and strong temperature gradients cause strongly negative values of $E_r$. The deep well of $E_r$ (and large $\nabla E_r$) extends over a wide range from the plasma edge towards the center.

A general decrease of turbulence level is observed during the extended transition phase and neoclassical transport—though reduced by the electric field—dominates for $r < 2/3a$. Measured $E_r/r$ are close to those calculated from the neoclassical ambipolarity condition via the DKES code. The steep pressure gradient does not reside at the very plasma edge as in the conventional quiescent H-mode but is shifted to about $2/3a$. Small ELMs are also observed in H-NBI with a pivot point $r_{pp}$, which is located at the maximum of $-\nabla p$ and which has moved with it further in (from $r_s - r_{pp} = 1.5$ cm in H* to $\approx 4.5$ cm in H-NBI; $r/a_{\text{eff}} \approx 0.75$).

A prerequisite to achieve the H-NBI mode is sufficient ion heating at a low edge density and limiter operation preferably with the inner limiters. A minimum ion heating power is required; with reduced NBI the transition is delayed. The narrow density profile requires low recycling conditions (good wall conditioning by boronization) and preferentially beam fueling from NBI (historically the latter gave the name for this regime). Strong gas puffing and ECRH—both increase the edge density—can lead to a destruction of the H-NBI mode.

Under controlled conditions, the H-NBI mode can also develop with ECRH or ECRH plus NBI even under separatrix conditions around $\nu = 0.5$. There is, however, the latent tendency that the maximum in $|E_r|$ decreases and moves to the plasma edge and the discharge develops to a conventional ELMy H-mode without further improved confinement properties.

The best confinement times of W7-AS ($\tau_E = 60$ ms) have been observed in the H-NBI regime with moderate heating power $P_{\text{NBI}} = 0.35$ MW and maximum density which could be obtained stationary under these limiter conditions. At moderate densities, $(n_e) = 0.6 \times 10^{20}$ m⁻³, and maximally achieved heating power $(P_{\text{NBI,abs}} = 0.85$ MW + $P_{\text{ECRH}} = 0.35$ MW) high values of $T_i$ and of $n_eT_i\tau_E$ have been obtained (see Table II).

VIII. STABILITY

In stellarator operation proper, only pressure driven instabilities develop in W7-AS: MHD modes, which are driven by the thermal pressure, and global Alfvén eigenmodes (GAE), which are driven by the fast particle pressure of NBI. At the edge, ELMs appear in the H-mode, destabilized by either the edge pressure gradient or the edge bootstrap current. Only with additional currents, driven ohmically or with ECCD, current driven instabilities are induced.

Pressure driven MHD does not play a detrimental role as long as the location of the resonance does not reside in the region of steep gradients. High-$\beta$ discharges are found to be rather quiescent, because both MHD and GAE modes are stabilized (see Fig. 14 and the related discussion). Figure 29 shows measured and computed (with \textsc{cas}3d\textsuperscript{69}) GAE $m=3$ mode patterns. The GAE-mode is localized just beneath the (3,1) shear Alfvén continuum band. Only bursting high-frequency GAE modes lead to beam particle losses or beam-power redistribution. Detailed reports are available on the GAE modes observed on W7-AS\textsuperscript{70,71} a detailed account of MHD studies on W7-AS is given in Ref. 71.

ELMs in W7-AS have basically the same features (relative location, duration, broadband frequency spectrum, energy loss, level of power deposition onto target plates) as in
TABLE II. Maximal plasma parameters of W7-AS.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$T_e$ = 6.8 keV</td>
<td>$B_0$ = 2.5 T; $\iota_e = 0.34; n_e = 2 \times 10^{19} \text{ m}^{-3}; P_{\text{EBRH}} = 2 \text{ MW}; $ electron root</td>
</tr>
<tr>
<td>$T_i$ = 1.7 keV</td>
<td>$B_0$ = 2.5 T; $\iota_i = 0.52; n_e = 5 \times 10^{19} \text{ m}^{-3}; P_{\text{EBRH}} = 0.5 \text{ MW}; P_{\text{NBI}} = 1.3 \text{ MW}; $ ion root; NBI-H-mode</td>
</tr>
<tr>
<td>$n_e$ = $4 \times 10^{20} \text{ m}^{-3}$</td>
<td>$B_0$ = 2.5 T; $\iota_e = 0.55; n_e = 2 \times 10^{20} \text{ m}^{-3}; P_{\text{NBI}} = 2.4 \text{ MW}; $ HDH-mode</td>
</tr>
<tr>
<td>$\langle \beta \rangle$ = 3.4%</td>
<td>$B_0$ = 0.9 T; $\iota_e = 0.5; n_e = 2 \times 10^{20} \text{ m}^{-3}; P_{\text{NBI}} = 2.8 \text{ MW}; $ HDH-mode</td>
</tr>
<tr>
<td>$\tau_E$ = 0.06 s</td>
<td>$B_0$ = 2.5 T; $\iota_e = 0.345; n_e = 1.1 \times 10^{20} \text{ m}^{-3}; P_{\text{NBI}} = 0.33 \text{ MW}; $ H-NBI-mode</td>
</tr>
<tr>
<td>$n_e T_e R_E$ = $5 \times 10^{21} \text{ eV sm}^{-3}$</td>
<td>$B_0$ = 2.5 T; $\iota_e = 0.345; n_e = 1.1 \times 10^{20} \text{ m}^{-3}; P_{\text{NBI}} = 0.33 \text{ MW}; $ H-NBI-mode</td>
</tr>
</tbody>
</table>

between ELMs, there is a low power flux. Figure 30(a) shows the power flux onto the divertor target at the transition from an L-phase into an ELMy H-mode.

ELMs give rise to power fluxes up to 20 MW/m². Between ELMs, there is a low power flux. Figure 30(b) shows target plate deposition profiles for various conditions—L-phase, HDH-phase, H$^*$-phase, and the low flux between ELMs. For the cases investigated, the widths of the profiles are about the same; also the energy of an ELM is deposited at about the same profile width.

IX. OPERATIONAL LIMITS

A. Maximal parameters

The maximal parameters achieved with W7-AS under separate conditions are given in Table II.

B. High-density operation and density limit

High-density operation was carried out at W7-AS with gas puff fueling only (no multi-H-pellet injection was available). Successful high-density operation required good boundary conditions (e.g., the island divertor). Figure 31 shows the highest densities achieved with W7-AS compared with the density values stored in the ITER H-mode database. The two data groups are well separated: Unlike confinement, the physics of the density limit seems to be different between tokamaks and stellarators. In W7-AS, the density limit manifests itself by a gradual decay of the plasma owing to excessive edge radiation. At sufficient separatrix-target plate separation, the plasma entered into a stage of stable detachment. In this case, the effective plasma cross sections shrunk and the energy content reduced; closer to a limiter configuration, the plasma thermally collapsed. High density operation in W7-AS violated the power balance and not, like in tokamaks, an MHD stability condition. No disruptive density limit has been observed in W7-AS up to the measured $\beta$ values when operated as stellarator proper. The density limit of W7-AS follows the scaling: $n_e \text{max} = 1.462(P/V)^{0.48}B^{0.54}$ (MW, T, $10^{20} \text{ m}^{-3}$).

In case of tokamaks, plasma transport increases when the density limit is approached. In the case of W7-AS, the situation is different, as shown in Fig. 32. Plotted is the confinement enhancement factor $\ell$—the ratio of measured to scaled $\tau_E$ values—against line averaged density. The curves show cases with different effective radii, varied by small changes of iota. Thus, also the distance separatrix–target plate has been varied. The two cases with large $a_{\text{eff}}$ and small distance end in a rapid collapse caused by excessive impurity radiation; the three cases with smaller radii and larger separatrix–target distances transit into stably detached phases (right vertical bar). Plotted is also the Greenwald density limit calculated from the effective plasma current and the density limit values expected from the stellarator scaling as given above for the two extreme $a_{\text{eff}}$ values.

Close to the Greenwald limit, plasma confinement does not decrease like in tokamaks, but it increases at the transition to the HDH regime (left vertical bar). At the transition into detachment, confinement decreases. To what extent the decrease of the energy content is caused by the reduced plasma cross section or by an additionally enhanced transport could not be explored. Heat-wave studies employing the OXB-heating scheme at $n_e = 3.5 \times 10^{20} \text{ m}^{-3}$ did not show any anomalously enhanced $\chi_{\text{eff}}$. 

FIG. 29. (Color). $m = 3$ GAE mode at 28 kHz measured via SX tomography: (a) Alfvén continuum bands with the mode location in space and frequency; (b) measured mode pattern; (c) mode pattern calculated with the CASID code.
C. High-\(\beta\) studies

The maximal \(\langle \beta \rangle\) values in W7-AS of up to 3.4% were still power and not MHD limited with basically the same power degradation of \(\tau_E\) close to the \(\beta\)-limit as at low \(\beta\). The maximal core \(\beta\) values reached up to 7%. Maximal \(\beta\) values were obtained with beam heating in the range \(0.9 \, \text{T} < B_0 < 1.25 \, \text{T}\) (for lower field values, the effective beam heating power decreased). The collisionality of the core of the high \(\beta\) plasma is marginally in the plateau regime. Figure 33 shows the maximal \(\langle \beta \rangle\) values in comparison to those of the ITER database. The two data groups are again well separated. Like the density limit and different to confinement, also the mechanisms which limit \(\langle \beta \rangle\) seem to be different in W7-AS and tokamaks. A detailed account of high-\(\beta\) studies on W7-AS is given in Ref. 22.

X. STEADY-STATE OPERATION

Unlike W7-X, W7-AS is not equipped with superconducting coils. The potential for steady-state operation could only be demonstrated in pulses up to 1.8 s and from the characteristics of the impurity transport. Figure 34 shows a quasisteady-state HDH-discharge at high \(\beta\) with the time axes (abscissa) scaled by the confinement time during the plateau phase. The high-\(\beta\) phase can be maintained stably and quasisteady state for about 65 confinement times. [At the end of this discharge, the heating power increase (by about 200 kW) allows to assess a remaining response of the plasma and the proximity to the \(\beta\) limit.] The plasma parameters under these conditions are of high programmatic relevance: The SOL midplane density is typically \(3 – 8 \times 10^{19} \, \text{m}^{-3}\); the neutral pressure inside the divertor is about \(10^{-3} \, \text{mbar}\), sufficient for particle control. The ratio of impurity to energy confinement time can be as small as 2, which fulfills the general target for ignition (\(< 5\)); the confinement time (6 ms) agrees with the superior W7-AS scaling. To allow comparison with tokamaks: \(q = 2\), the Greenwald factor \(n/n_{GW} = 3.0; \beta_N = 12\).

Figure 35 plots the high-\(\beta\) values of W7-AS against the duration of the high-\(\beta\) phase and compares the results with...
those accumulated from tokamaks. Though W7-AS did not reach the spectacular but transient beta values of tokamaks, it is able to maintain high $\beta$ values for longer phases than achieved in tokamaks (status 2003). This quality may already indicate the intrinsic property of stellarators for steady-state operation.

XI. $n_e T_i E$ VALUES

Figure 36 shows selected $n_e T_i E$ data points of W7-AS, again achieved at operational boundaries against the magnetic energy ($B_0^2$ Vol). The data points are compared with tokamak data. The small W7-AS device does, of course, not reach the high values of the large tokamaks. But the results lie amidst the tokamak data though they are obtained close to operational boundaries (beta, density, confinement).

The data point for W7-X at the maximal density and heating power, expected along the W7-AS confinement and density limit scalings and W7-AS profile shapes, is also shown.

XII. THE NEXT STEP: W7-X

After nearly 14 years with 56,953 discharges, W7-AS suspended operation at the end of July 2002. The development of the Wendelstein stellarator line will continue with the W7-X device. W7-X, presently under construction in Greifswald, Mecklenburg-Vorpommern, Germany, is a fully optimized stellarator. Its optimization is based on the concept of quasi-isodynamicity. A truly isodynamic confinement geometry has poloidal symmetry with the consequence that plasma flow occurs only on flux surfaces without radial contributions. Such a system would avoid the orbit losses of fast particles and the radial neoclassical fluxes would be zero. In toroidal geometry, only quasi-isodynamic symmetry can be realized.

W7-X has, as its predecessors W7-A and W7-AS, a five-fold toroidal symmetry. The magnetic field is increased at the corners of the pentagon to localize reflected particles to the straight sectors where their banana orbits rotate poloidally. This linked-mirror concept excludes trapped particles from the zones of strong field inhomogeneity. A quasi-isodynamic system allows the same neoclassical particle confinement quality as given hypothetically in true isodynamicity or with axisymmetry. Figure 37 compares the orbits of a trapped particle in the W7-AS and the W7-X geometry: in W7-AS, it is on a loss orbit, in W7-X it precesses poloidally.

In order to provide flux surfaces with small Shafranov shift toward high $\beta$, the pressure driven Pfirsch–Schlüter currents have to be minimized. Quasi-isodynamicity allows the reduction of $\langle j^2 \rangle / \langle i^2 \rangle$, which is $2 / i^2$ for the classical stellarator to $0.32 / i^2$ for W7-X ($0.85 / i^2$ for W7-AS). Improved equilibrium conditions are achieved by strong shaping; the elongation of W7-X is 3.6 in the corners of the pentagon. Figure 38 compares the pressure driven current lines of...
W7-AS with those of W7-X. The optimization achieved with W7-X is obvious; the current flows mostly perpendicularly to the field lines.

For the goal of a flux surface geometry which is rather independent of plasma pressure, also the bootstrap current was minimized for W7-X. This was possible in the frame of quasi-isodynamicity by relating the helical and toroidal field Fourier coefficients. The rotational transform of W7-X has been selected close to \( i = 1 \). Two limits are possible with \( i \) varying from core to edge between 0.8, \( i \), and 1.2. Thus maximal shear is possible between these boundaries whereas the major resonances 5/6, 5/5, and 5/4 are avoided. An edge separatrix is formed either with the 5/5 or the 5/4 island chains. Like on W7-AS they will be used to operate W7-X with an island divertor.

W7-X is optimized according to the following criteria:

(a) nested flux surfaces without major resonances and only small islands;
(b) low Shafranov shift toward high \( \beta \);
(c) good MHD stability properties with a stability limit close to \( \langle \beta \rangle \approx 5\% \);
(d) low neoclassical losses in the long-mean-free-path regime;
(e) low bootstrap current in the long-mean-free path regime to maintain the field optimization from low to high \( \beta \);
(f) sufficiently long \( \alpha \)-particle confinement time in a later reactor \( (\tau_\alpha \approx \tau_{\text{slowing down}}) \).

Figure 39 shows computer graphics of W7-X with plasma, divertor installation, the plasma vessel, the two sets of superconducting coils (nonplanar modular coils and planar coils to change, e.g., rotational transform), the outer cryostat. The major design parameters of W7-X are shown in Table III.

The final figures show the development of W7-X (status end of 2004). Figure 40 shows the first modular coil in the manipulator with which it will be moved during assembly over the vessel onto its final position; Fig. 41 shows the same for a planar coil; Fig. 42 shows a photo of the vessel piece partly covered with the thermal insulation; and Fig. 43 shows the Maquette 140 GHz steady-state gyrotron as operated in Greifswald.
XIII. CONCLUSIONS

The Wendelstein stellarator line has been developed as large aspect ratio, low shear concept, originally employing helical coils. With W7-AS for the first time in a large fusion device, copper modular coils were used. This choice is necessary to overcome the technical limitations of helical coil systems in large stellarators. Modular coils provide the technical flexibility to build optimized stellarators following different optimization principles. For W7-X, the first superconducting modular coils have been built under industrial series production conditions.

The last device of the Wendelstein stellarator line in operation, W7-AS, has fulfilled its major goals: to test the modular coil concept; to develop the island divertor; to test the effectiveness of the first steps in stellarator optimization and thus to contribute to the scientific basis for W7-X; and to contribute to the physics of toroidal confinement and specifically to stellarator physics.

In W7-AS divertor operation was possible with good plasma confinement under quasisteady-state conditions in an ELM-free variant of the H-mode, the HDH-mode, accessible at high densities. The H-mode power threshold is low. Operation at high densities well above the Greenwald limit was possible. The heating power limited $\beta$ values were found to be high. The plasma performance and confinement quality

<table>
<thead>
<tr>
<th>TABLE III. Major design parameters of W7-X.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parameter</td>
</tr>
<tr>
<td>Major radius</td>
</tr>
<tr>
<td>Minor radius</td>
</tr>
<tr>
<td>Plasma volume</td>
</tr>
<tr>
<td>Nonplanar coils</td>
</tr>
<tr>
<td>Planar coils</td>
</tr>
<tr>
<td>Induction on axis</td>
</tr>
<tr>
<td>Stored magnetic energy</td>
</tr>
<tr>
<td>ECRF heating power</td>
</tr>
<tr>
<td>NBI heating power</td>
</tr>
<tr>
<td>ICRH heating power</td>
</tr>
<tr>
<td>Nominal pulse length</td>
</tr>
<tr>
<td>Machine height</td>
</tr>
<tr>
<td>Machine diameter</td>
</tr>
<tr>
<td>Machine mass</td>
</tr>
<tr>
<td>Cold mass</td>
</tr>
</tbody>
</table>

FIG. 40. (Color). Shown is a W7-X modular coil mounted into the manipulator, which is used to string the coil over the plasma vessel during assembly.

FIG. 41. (Color). Shown is a W7-X planar coil mounted into the manipulator, which is used to string the coil over the modular coils during assembly.

FIG. 42. (Color). Semi-module of the plasma vessel partly covered with the thermal insulation.
does not noticeably degrade close to limits. Also close to operational limits, the plasma performance is found to be free of violent instabilities. Experimentally, the development of Wendelstein stellarators will continue in 2010 with W7-X operation. W7-AS operation and W7-X design have contributed to further establish the Wendelstein stellarator concept as an independent reactor line.

ACKNOWLEDGMENT

The operation of W7-AS was under a high time pressure specifically in the final years. Without the high motivation and the high standard of technical work of the W7-AS operation and diagnostic technical teams we would not have accomplished what is presented in this paper.

1L. Spitzer, Jr., Phys. Fluids 1, 253 (1958).


14Y. Feng, F. Sardei, P. Grigull et al., Proceedings of the 39th EPS Conference on Controlled Fusion and Plasma Physics, St. Petersburg, 2003 (European Physical Society, Mulhouse, 2003), Vol. 27A, p. 4-4.4-C.


20D. A. Hartmann, “Ion cyclotron range of frequency experiments on the stellarator W7-AS,” Nucl. Fusion (to be published).


