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From Wendelstein 7-X to a Stellarator Reactor

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Wendelstein 7-X is a drift optimized stellarator with improved thermal and fast ion confinement. Additional optimization criteria are a stiff equilibrium configuration and MHD stability up to a volume averaged β of 5%. The main objectives are to demonstrate reactor relevant plasma performance under steady state conditions including power and particle exhaust with an island divertor. To that effect Wendelstein 7-X has superconducting coils with a maximum average magnetic field of 3 T and will be equipped with actively cooled plasma facing components for heat fluxes of up to 10 MW/m^2. Besides fulfilling this research mission, the extrapolation from Wendelstein 7-X to a stellarator reactor will depend on the comparison with the results from other stellarators, the ITER results and here in particular the experience gained with α-particle heating and operating a nuclear device, and the possibility to extrapolate these results by first principle theory.

Keywords: Wendelstein 7-X, stellarator optimization, Helically Advanced Stellarator

1. Introduction

Wendelstein 7-X (W7-X) is a drift optimized low magnetic shear stellarator to demonstrate basic reactor capability of the stellarator concept [1]. With a major radius, R, of 5.5 m and an average minor radius, <a>, of 0.5 m the resulting plasma volume of 30 m^3 lies between those of ASDEX Upgrade and JET. The maximum magnetic field is 3 T, corresponding to 600 MJ of magnetic field energy. The rotational transform, ι, ranges from 5/6 to 5/4 and, in contrast to the partially optimized predecessor of W7-X, Wendelstein 7-AS (W7-AS) [2], is practically independent of the plasma β.

The design of W7-X is based on an elaborate optimization procedure to overcome the essential deficiencies of the stellarator concept: (1) The introduction of quasi-symmetry – in case of W7-X a quasi-isodynamic configuration has been chosen – yields reduced neoclassical transport and, in particular, good fast ion confinement which is a prerequisite for any type of fusion reactor. Since in a stellarator the neoclassical diffusion scales like $\varepsilon_h T^{7/2}$, the helical ripple, ε_h, has to be kept as small as possible. (2) By minimizing the Pfirsch-Schlüter and bootstrap currents the Shafranov shift is minimized and thus a high equilibrium limit is achieved. (3) Finally, the magnetic field configuration provides sufficiently large magnetic well to avoid pressure drive instabilities such as interchange modes, aiming at an volume averaged $\langle\beta\rangle$ of 5%.

High power steady state operation of W7-X will be approached in two steps: (1) An inertially cooled test divertor will allow pulses from 10 to 50 s, corresponding to heating power levels from 8 to 1 MW. During this initial phase three heating systems will be available: Neutral beam injection (NBI), ion cyclotron resonance heating (ICRH) or electron cyclotron resonance heating (ECRH). Depending on the combination of the heating methods, the available heating power will vary between 8 and 11 MW. (2) After the installation of the actively cooled divertor and the completion of the water cooling of all plasma facing components, 30 minutes plasmas with 10 MW ECRH [3] are foreseen. Later upgrades will include increases of the neutral beam heating power from 10 to 20 MW and of the ion cyclotron heating power from 2 to 10 MW.

Various studies of a HElical Advanced Stellarator (HELIAS) reactor have been conducted already (see e.g. [4,5]). The HELIAS reactor is basically an extrapolation from the W7-X design, which in itself is based on results from the W7-AS stellarator, the first advanced stellarator experiment. Essentially, three requirements form the basis of the HELIAS reactor concept: (1) Sufficiently good confinement has to be guaranteed to reach ignition. Probably owing to the different weighting of neoclassical and anomalous transport in the different stellarator devices, a unified scaling law does not exist. Recent studies have shown that the cross-device scaling improves with the introduction of a configuration factor, which appears to correlate with the degree of neoclassical optimization [6]. Therefore, conservatively an confinement improvement, such as observed in the H-mode, has not been assumed. (2) To provide a super-conductor with sufficient ductility for the fabrication of the non-planar modular coils, NbTi has been chosen. At the temperature of super-fluid helium of 1.8 K maximum magnetic fields of 10 T are possible, corresponding to 5 T on axis. (3) For the blanket a space

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with 1.3 m thickness has been reserved between plasma and coils. Although no detailed design for the blanket has been made up to now, this is thought to be sufficient. With $<\beta>$ = 4 – 5%, the resulting HELIAS reactor concepts have 3 GW of fusion power. Three types with different aspect ratios, but similar plasma volumes, have been investigated [7]: The HR22/5 with a five-fold symmetry, a major radius of R = 22 m, an aspect ratio of A = 12 and 50 modular coils, the HR18/4 with R = 18 m, A = 9 and 40 modular coils, and the HR3/15 with R = 15 m, A = 6 and 30 modular coils. Less coils for a given plasma volume means lower cost. However, the limit for reducing the aspect ratio is given by the α-particle losses, which for the HR3/15 already become critically high. Therefore, the HR18/4 is regarded as the best compromise between a reasonably small number of coils and sufficient α-confinement. Because of the low volume to surface ratio, the advantage of a large aspect ratio is a low neutron flux to the wall which for the HR4/18 is calculated to be on average 1 MW/m^2 reaching peak values of 1.6 MW/m^2.

Recently the European fusion facilities have undergone an extensive review to assess their relevance for the future fusion programme. Seven R&D missions have been defined which provide an efficient and focused implementation of the fusion programme [8]. Their titles are burning plasma, reliable operation, operation compatible with first wall, technology and physics for steady state operation, predicting fusion performance, operation in nuclear environment and DEMO integrated design. The following discussion of the W7-X programme will be made with respect to these R&D missions. The last two missions, however, do not apply, as W7-X is neither a nuclear device (no deuterium-tritium operation) nor will DEMO components such as a breeder blanket be tested in W7-X.

2. Burning Plasma

The confinement of the fast helium ions or α-particles from the D-T fusion reactor is a prerequisite for a future fusion reactors. While in tokamaks it has been demonstrated that at least for low fast ion pressure α-particles are confined [9] and heat the plasma [10], in stellarators this is not so easily achieved. Without optimization of the stellarator magnetic field configuration, in the long mean-free path regime fast ions tend to drift radially and thus leave the confinement region. The quasi-isodynamic symmetry of W7-X solves this problem by increasing the magnetic field in transition areas between the five field periods, basically establishing a system of linked mirrors [11]. The trapped particles oscillate between these regions of high magnetic field, making a net poloidal rotation but no radial movement as they are kept away from zones of high field inhomogeneity. Fig. 1 shows the orbits of 50 keV protons calculated for W7-AS and W7-X. It can be seen that for the partially drift-optimized W7-AS the fast ions are not confined for all pitch angles.

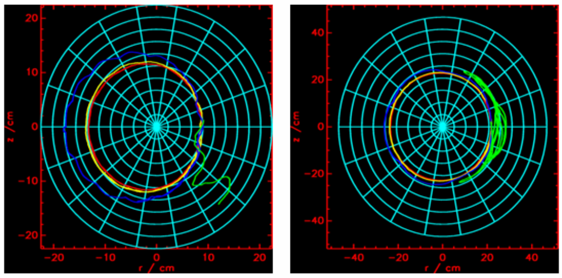

Fig. 1 Orbits of 50 keV protons in the poloidal projection for W7-AS (left) and W7-X (right). The colors denote the pitch angle: 0°, 50°, 70°, 80°

A particular characteristic of the W7-X configuration is that the collisionless fast particle confinement requires a finite β of about 2–3% [12]. As a consequence, full demonstration of fast ion confinement will be possible only during the 2nd phase of operation, as power and pulse duration are limited during the 1st phase. To reach a fully equilibrated magnetic field configuration the pulse duration has to be of the order of $\tau = L/R \approx 20$ s.

Extrapolated to the HELIAS reactor (HR4/18) a α-loss fraction of 2.5% is predicted. With respect to the power balance this is in any case not critical, but in a fusion reactor the α-losses have to be kept this low, as undue localized fast ion fluxes leaving the plasma damage first wall components.

3. Reliable Operation

To achieve a high fusion power density, a fusion reactor requires high β and high plasma density. This, however means, that reliable operation near operational boundaries will be needed. Here, stellarators have a clear advantage as, without current driven instabilities and disruptions, the plasma behaviour at the β-limit is fairly benign. In addition, much higher densities can be achieved in stellarators as the Greenwald limit, known from tokamaks, has not been observed [2].

While W7-AS still showed pressure driven modes, the W7-X design should include sufficient magnetic well to provide stability up to $<\beta>$ = 5%, at least for the standard and high mirror configurations. However, also in W7-AS examples exist where the increase of $<\beta>$ eventually led to a stabilization of these modes. This is explained by the formation of a magnetic well as the configuration changes with rising $<\beta>$. Owing to a small Shafranov shift also to equilibrium limit of W7-X should not curtail the value of $<\beta>$ = 5%.

Already during the 1st operational phase high density will be addressed. To this effect W7-X is equipped with an ECRH system prepared for 2nd harmonic O-mode which works above electron densities of $n_e = 1\times10^{20}\ m^{-3}$. Based on neoclassical transport in the plasma core (and an anomalous edge) at a density of $1.8\times10^{20} m^{-3}$ and 10 MW of ECRH, electron and ion temperatures of $T_e = 6.2$ keV and $T_i = 4.2$ keV, corresponding to $<\beta> = 4.1\%$, are predicted. Depending on the actual confinement, high β might be possible only at lower magnetic field (meaning below 2.5 T, which is the nominal field for 140 GHz ECRH). During the 2nd phase of operation β- and equilibrium limit studies will become possible at power levels of 20 MW and above.

With $<\beta> = 4$–5% and $n_e = 2\text{-}3\times10^{20}\ m^{-3}$ the HELIAS does not exceed the values envisaged for W7-X. Confinement time (1.6 – 2.3 s) and ion temperature ($T_i = 11 – 15$ keV) are of course higher.

4. Operation Compatible with First Wall

On the one hand, an undue contamination of plasma with impurities from the first wall has to be avoided and, on the other hand, wall erosion and tritium retention have to be limited. Therefore, carbon is ruled out as a first wall material and tungsten is considered as a candidate material for a fusion reactor [13].

Nevertheless, W7-X will start operation with carbon covering the high heat flux target elements ($\geq 1\ MW/m^2$). For stellarators in particular, because of effectively missing temperature screening effect, impurity accumulation is a critical issue. For the power and particle exhaust W7-X is equipped with an island divertor which utilizes the large magnetic islands forming at the plasma boundary at $\iota = 1$. This concept was for the first time successfully tested in W7-AS. In W7-AS the introduction of the island divertor also led to the discovery of the high density H-mode (HDH-mode), which not only showed improved energy confinement, but at the same time much reduced impurity confinement [14]. This very favourable behaviour is illustrated in Fig. 2. The explanation for the low impurity confinement are strong density gradients at the plasma edge, which hinders the penetration of impurities, and an additional outward impurity transport. The latter, however, is not understood. Because of these properties the HDH-mode is the candidate scenario for high density steady state plasma operation in W7-X. However it is unclear how the HDH scales to W7-X.

During the 1st phase of W7-X operation first the divertor topology will have to be investigated. Here, the inertially cooled test divertor has the advantage that, because of its intrinsically robust design, overheating or damaging the cooling structure does not have to be considered. Subsequently, first attempts to re-establish the HDH-mode will be made. The 2nd phase of operation will then address full steady state power and particle exhaust. Eventually also high-Z wall materials, such as tungsten, will have to be considered. Here, a gradual increase of the wall coverage as successfully demonstrated in ASDEX Upgrade [15] could be a feasible approach.

5. Technology and Physics for Steady State

W7-X is the first optimized stellarator with an integrated design for steady state operation. This includes super-conducting coils made of NbTi for a magnet field of up to 3 T on axis, an actively cooled first wall for heat fluxes of up to 10 MW/m² [16], and device control, data acquisition, diagnostics and an ECRH system [3] developed for continuous plasma operation. Many aspects of these technologies are similar to the ITER requirements.

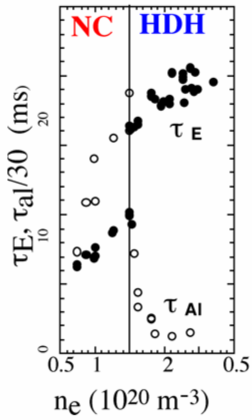

Fig. 2 Energy and impurity confinement time as a function of plasma density. The transition from normal confinement, NC, to the HDH-mode becomes evident in a rise of τ_E and a simultaneous drop of τ_{al} (which is the measured decay time of aluminum injected by laser blow-off).

During the 1st operational phase of W7-X stationary plasma operation is only possible at very low power levels of the order of 100 kW to test basic system properties. Going to 1 MW, 50 s pulse will already give some insight into plasma behaviour for times longer than the *L/R* time. In addition to what is described in the previous chapters, short pulses (8 MW for ~10 s) will be used to verify the improved neoclassical transport. In the 2nd phase the objective is to develop a fully steady state high power plasma scenario demonstrating the reactor capability of the stellarator.

The plasma control requirements of W7-X are generally very low. This also applies to the necessity to control the magnetic field. One of the optimization criteria of the W7-X design is the minimization of the bootstrap current. While the smallest bootstrap current is expected in the high mirror configuration, calculations show that depending on confinement and density bootstrap currents

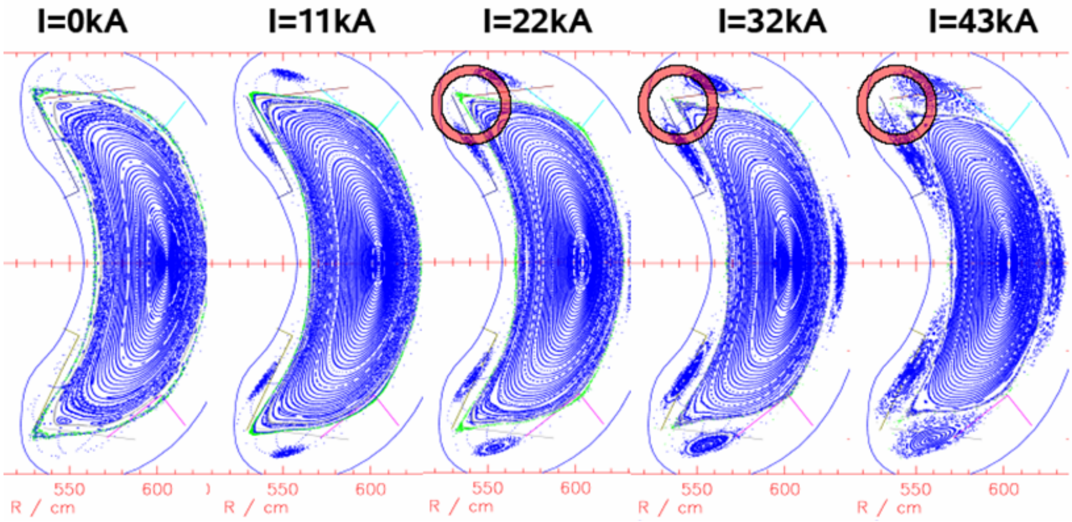

Fig. 3 Sequence of W7-X equilibria with different amounts of plasma currents simulating the current diffusion while a bootstrap current of 43 kA is building up. The red circles show the critical regions near the edges of the divertor tiles.

of up to 50 kA might develop at 5 MW heating power. This level is thought to be adequate to go from 2nd harmonic X-mode to 2nd harmonic O-mode, as in O-mode the microwave absorption strongly increases with electron temperature and a thermal collapse has to be avoided [17]. As a consequence of the bootstrap current the plasma equilibrium will show an initial temporal evolution. To illustrate this effect, a sequence of equilibria with different amounts of toroidal currents have been calculated using the VMEC code and a field extender for the magnetic island regions outside the last closed flux surface (Fig. 3). The current diffusion is simulated by assuming that the bootstrap current, I_{BS}, is initially balanced by an ohmic current, $\int \sigma E\, dA$ (E: toroidal electric field, $\int dA$: integral over poloidal cross-section), where the electrical conductivity, σ, is essentially given by the temperature

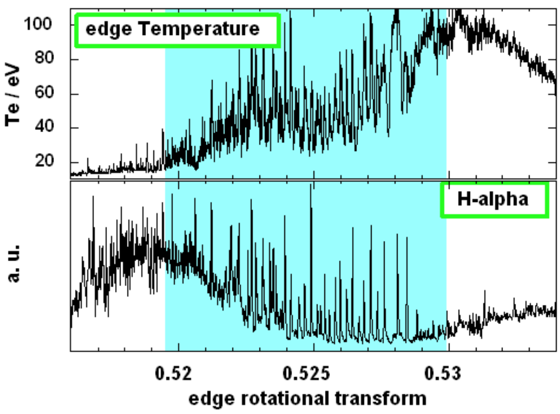

Fig. 4 Variation of the edge electron temperature and the ELM characteristic (from Hα signal) with edge rotational transform in W7-AS [2].

distribution. The ohmic current is then successively reduced, until finally the total current, $I = I_{BS} + \int \sigma E\, dA$ matches the bootstrap current. Fig. 3 shows an example where the final equilibrium with the fully developed plasma current is consistent with the island divertor operation. However, on the way to this configuration, the edges of the divertor tiles and the pumping gap will be loaded with plasma. To avoid high thermal loads at these regions, either current drive has to be applied simultaneous with ECRH or the rotational transform has to be adjusted. In W7-X the latter should be in principle possible by using the planar coils, but is technically limited due to a limited number of allowed load cycles of the magnets. An additional option is the introduction of protection limiters.

6. Predicting fusion performance

The prime objective of W7-X is to demonstrate the basics reactor capability of the stellarator. W7-X will have to verify the theory based optimization criteria and thereby also improve the theoretical understanding of stellarator physics. In a more general context, there are however also a range of physics issues related to 3D effects concerning both stellarators and tokamaks. E.g., in stellarators the generation of ambipolar electric are caused by the open magnet field lines in the plasma boundary. Recently, the effect of such electric fields on momentum transport has been observed in tokamaks with an ergodic divertor [18]. The control of edge localized modes (ELMs) using dedicated perturbation coils to ergodize the plasma edge is now one of the methods to mitigate ELMs in ITER. In this context W7-X could provide information on ELM control without strong plasma currents and, thus, without the contribution of current driven instabilities. As an example Fig. 4 shows the effect of the change of the rotational transform on the ELM signature in W7-AS [2].

The size of the step from W7-X to a larger stellarator device towards a stellarator reactor will depend on the ability to extrapolate the results. This includes the transferability of the ITER results to stellarators, and here in particular the α-particle heating, and will strongly depend on the progress made with first principle theories and their applicability to 3D magnetic field configurations. In this context Wendelstein 7-X fits well into the ITER schedule and the development plan for high performance computing.

7. Summary and Conclusions

Wendelstein 7-X is an optimized stellarator designed for steady state operation. Its main objective is the demonstration of the basic reactor capability of the stellarator.

W7-X addresses the main physics issues for the development of a stellarator reactor: (1) Fast particle confinement and fast particle driven instabilities in a 3D configuration. Owing to the high plasma densities a reduced drive is expected for the latter. (2) Neoclassical versus turbulent transport. Latest calculations suggest also for the turbulent transport a dependence on the degree of neoclassical optimization [19]. (3) Impurity confinement. Plasma scenarios, such as the HDH-mode have to be further developed to avoid impurity accumulation. Also here the possible role of a turbulent drive has to be investigated. (4) 3D divertor configuration. W7-X will utilize the natural magnetic islands at the plasma boundary and combine them with actively cooled divertor targets.

Correspondingly specific stellarator technology issues are: (1) Coil configuration, coil support structure and the choice of superconductor. In a stellarator with modular non-planar coils the forces on the support structure need particular attention. (2) Divertor. In a 3D magnetic field configuration a divertor is also technically more complicated, requiring very accurate alignment to avoid unbalanced heat load distributions. (3) Depending on the aspect ratio, the possibility to provide enough space can be very limited in stellarators. (4) In addition, both accessibility and maintainability need to be thoroughly investigated in future stellarator reactor studies.

The further optimization of stellarators has to include also simplified engineering solutions. To that effect a first-of-a-kind device such as W7-X is not optimized. Future stellarator reactor studies will have to combine in a more rigorous way physics and engineering optimization.

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