Island Divertors: Concepts and Status of Experimental and Modelling Results

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Abstract. Basic features of the island divertor concepts for low-shear advanced stellarators like W7-AS and W7-X (intrinsic island divertors) and for high-shear heliotrons like CHS and LHD (local island divertors, LID) and first results for both concepts are shortly reviewed. The diverting fields of island divertors are either intrinsic (low-shear case) or externally imposed (high-shear case). The associated field perturbations are very small compared to tokamak poloidal field divertors, which explains the high flexibility of island divertor configurations, but sufficiently large to generate divertor-viable islands. Although the physics of island divertors is expected to be similar to that of tokamak poloidal field divertors, leading geometrical parameters significantly differ from those of comparable-size tokamaks. Furthermore, strong three-dimensional effects arise from toroidally discontinuous target plates. For both the intrinsic and local island divertor configurations, the island structure and the plasma diversion have been verified experimentally. For W7-AS, stable high recycling conditions could be demonstrated in a stellarator for a simple island divertor geometry. In CHS with the LID field switched on, a strong increase of the pumping efficiency was measured, resulting in a 50% reduction of the core density for the same gas-puff rate as in the non-LID case. 3D numerical transport studies with the EMC3/EIRENE code predict, for typical W7-AS values of the upstream density and power into the SOL, partial detachment with 80% power loss by carbon radiation in the SOL and significant momentum losses associated with the island divertor geometry.

1. Introduction

The island divertor (ID) concept is more than 25 years old and was first proposed for tokamaks as a small-current alternative to the poloidal field divertor, the basic idea being that magnetic islands can be obtained from arbitrarily small cross-field resonant perturbations of rational field lines [1,2]. Ten years later the concept was proposed for the ι = 5/9 island configuration of W7-AS [3,4] in the two model versions of an open and a closed divertor. In the past few years theoretical and experimental efforts have been considerably increased to explore the potential of the ID concept for solving the exhaust problem in helical devices [5-13].

Although the physics of IDs is expected to be similar to that of tokamak divertors, leading geometrical parameters are significantly different. For example, W7-AS and W7-X have a field-line pitch of 0.001-0.005 compared to ≈ 0.1 for tokamaks. This explains why L_c for IDs (≈ 80 m for W7-AS) is generally larger than L_c for even large tokamaks. The distance between the main plasma and the target plates is also much smaller than in tokamaks, leading to a less efficient screening of the recycling neutrals. Both a larger L_c and a smaller plasma-to-target distance increase the role of the cross-field transport. A further difference is the inherent three-dimensionality of IDs. It arises from the discontinuity of target plates, which implies a toroidal localisation of the recycling neutrals and ionisation sources, and from the toroidal variation of the island radial width, which implies a variation of the radial transport fluxes along the islands. Both effects lead, for temperatures < 10 eV, to a periodic modulation of the plasma parameters along the island tube, which cannot be smoothed out by parallel heat conduction [9].

2. Divertor islands for low-shear and high-shear configurations

Divertor islands originate from small radial field perturbations, $b_{mn} \equiv B_{mn}/B_{tor}$, resonant to low-order rational ι values at the boundary of the configuration, $\iota = m/n$. Since the radial island size decreases with the magnetic shear for a given field perturbation [14],

$$2\delta \propto \sqrt{b_{mn}/(\iota' n)} \quad , \tag{1}$$

the IDs of low-shear advanced stellarators (W7-AS and W7-X) can exploit large intrinsic island chains at the main edge resonance (ι_a=5/9 for W7-AS) [6,9,15], whereas the IDs of high-shear heliotrons (CHS and LHD) are based on large islands created by additional resonant field perturbations (ι_a=1 for LHD) [7,8,15]. In contrast to tokamaks, the field perturbations associated with IDs are very small ($\Delta B_{rad}/B_{tor}$ ~ 10^{-3}–10^{-4}), which means that additional resonant perturbation of the same magnitude from separate coils are sufficient to modify the island geometry.

In the low-shear case, the island region is topologically closed if the intrinsic "perturbation" field is sufficiently weak (W7-X, standard case). This field becomes larger by approaching the coils, i.e. by shifting the separatrix radially outwards or by raising the field of suitable ID control coils [13] or by plasma pressure effects [16], leading eventually to a break-up and ergodisation of the island structures beginning at the separatrix and gradually extending towards the O-point. However, for low-shear configurations like W7-AS and W7-X the ergodisation is sufficiently weak as to not destroy the basic field structure of the dominant ι_a = 5/n edge resonance.

A minimum island size for a useful ID can be estimated from the condition that // transport dominates over ⊥ transport. This is the case if δ is larger than the radial decay lengths of the plasma parameters from heat and particle balance [9]:

$$\delta > \lambda_T \equiv 7/2\, L_c \sqrt{\alpha \chi n_{up} / \kappa T_{up}^{5/2}} \quad , \tag{2}$$

$$\delta > \lambda_n \equiv \sqrt{D L_c / c_s} \quad , \tag{3}$$

with $\alpha \equiv L_X / L_c (1 - 0.5 L_X / L_c)$, L_X / L_c being the fraction of L_c facing the main plasma. The connection length is approximately

$$L_c \approx 2\pi R / n\Delta\iota, \quad \Delta\iota = \delta\iota' \quad . \tag{4}$$

Substitution of L_c into eqs. (2) and (3) gives

$$\delta > \delta_{min} = max\left((KD/c_s)^{1/3}, \left(7/2\, K \sqrt{\alpha\chi n_{up} / \kappa T_{up}^{5/2}} \right)^{1/2} \right) \quad , \tag{5}$$

with $K \equiv 2\pi R / n\iota'$. If $\delta < \delta_{min}$, the plasma reaches the targets via cross-field transport and limiter SOL conditions are approached. Eq. (2) shows that the competition between ⊥ and // heat transport is governed, for given plasma parameters, by the parameter δ / L_c. This geometric parameter is a figure of merit for the basic ID function and can be varied externally by appropriate control coils (see later). For W7-AS moderate density discharges at ι_a =5/9 with $\iota' \approx 0.5\ m^{-1}$, n = 9, $n_{up} = 3 \cdot 10^{19}\ m^{-3}$, T_{up} = 100 eV, χ = 3D=0.6 m^2/s, L_X / L_c = 0.7, eq. (5)

yields δ_{min} = 2.8 cm (from heat transport balance). Since δ = 2.5-3 cm, the diverting condition is only marginally satisfied, i.e. there is a considerable $\perp$ thermal transport across the islands.

3. Flexibility of island divertors

The weak sensitivity of the main configuration to the field perturbations associated with the islands implies a high flexibility of the ID geometry compared to that of tokamak divertors. For both low- and high-shear configurations the position, size and L_c of the islands can be varied with island control coils [13]. By varying the currents I_{cc} of the W7-AS control coils between -3.5 and +3.5 kA, which corresponds to a relative magnetic field variation of $\approx 10^{-3}$, configurations with large islands/small main plasma or small islands/large main plasma can be produced. A change of the plasma radius by up to 35% has been verified via the a^2 scaling of the measured W_{dia}. Since the island radial width δ scales as $1/L_c$ (eq. 4) and δ/L_c is the geometric parameter which governs the balance between // and $\perp$ heat transport in the ID, the control coils are an effective tool to investigate the diverting function of the ID. This effect has been verified by the measured power deposition profiles from IR camera observation. By reducing I_{cc} from –3.5 kA to 0, which corresponds to an increase of L_c from 60 to 110 m, the power deposition profile became broader by a factor of 1.6, reflecting the increasing weight of the $\perp$ transport [17]. Additionally for low-shear stellarators, the main edge resonance can be varied by the toroidal field coils. W7-X, for example, has three main edge resonances which can be used for an ID: the standard $\iota\!\!\!-$ (= 5/5) case, a low $\iota\!\!\!-$ (= 5/6) and a high $\iota\!\!\!-$ (= 5/4) case [18]. The island radial width is very similar, but the position of the last closed flux surface changes from behind the islands in the low $\iota\!\!\!-$ case to before the islands in the high $\iota\!\!\!-$ case. This reflects the higher ergodicity in the second case. The target and baffle geometries have been optimised to allow divertor operation at all three $\iota\!\!\!-$ values. This was achieved by shaping the targets so that in all three cases the estimated outflowing power is completely intercepted and the deposition profiles are as homogeneous as possible.

In Heliotron J, reasonable candidates for an ID are the island configurations $\iota\!\!\!-_a$ = 4/7 and 4/8 [19]. In the first case the islands are relatively small and there is no wall contact, since regular flux surfaces exist behind the islands. By raising the vertical field from the auxiliary coils, the $\iota\!\!\!-$ increases (second case), the main plasma size becomes smaller and the island larger, which leads, as already discussed, to smaller L_c. Here the radial field line diversion becomes larger and leads to wall contact. All these features are typical for low-shear IDs. By raising the vertical field further, the field line diversion becomes very large and the typical pattern of a helical divertor (HD) emerges. L_c is smaller than in the ID case due to the high radial field diversion.

High-shear devices like LHD can switch between helical and island configurations or use a combination of both after retracting the island divertor head [20]. The intrinsic HD can be viewed as the helical version of a double-null tokamak divertor except for the extended ergodic boundary ($\approx$ 5 cm). This ergodic region originates from a tight sequence of $\iota\!\!\!- = 10/n$ island layers which eventually overlap. The local island divertor (LID) is a closed ID with high pumping efficiency, favoured by the toroidal localisation of the recycling zone. The divertor island provides a sharp transition of about 2 mm from the closed to the strongly diverted region. In the third operational mode the HD and ID geometries are combined and the island directly connects the closed region to a strongly diverted stochastic boundary.

4. Main experimental results

The island structure, its integrity against moderate plasma pressure and the plasma diversion function have been verified experimentally for both low-shear and high-shear island divertor configurations by extensive plasma edge measurements [9].

For a W7-AS island configuration at $\iota = 5/9$, stable high recycling conditions were observed during a divertor pre-study campaign using a simple island divertor geometry [13]. In these high power, high gas puffing NBI experiments, the density at the separatrix, n_{es}, could be raised up to $7 \cdot 10^{19}$ m^{-3} after suppressing the H-mode with $P_{NBI} > 1.2$ MW. The density at the plates reached values of about $1.5 \cdot 10^{20}$ m^{-3}, showing a roughly quadratic increase with n_{es} , which is weaker than that resulting from the two-point model.

In CHS with the LID field switched on, the neutral pressure and the ion saturation current measured in the pumping duct are found to be a factor 1.5 times higher than without the LID field. As a result of the increased pumping efficiency by up to 10% in the first case, the line averaged core density is reduced by about 50%, although the gas puff is kept the same as in the non-LID case [21]. Additionally, a modest improvement of the energy confinement (20%) has been found with the LID.

5. Modelling results

A reliable prediction of the complex island divertor physics and a realistic interpretation of the experimental data have become possible by the development of the transport code EMC3 (**E**dge **M**onte **C**arlo **3**D) [12]. The standard 3D fluid-transport equations are written in a common Fokker-Planck form describing the conservation of particles, momentum, electron and ion energy during their convection and diffusion processes. The neutral code EIRENE is coupled selfconsistently. The EMC3 code uses a local field-aligned orthogonal coordinate system to represent the transport processes in parallel and perpendicular directions. In these coordinates the diffusion tensor is diagonal and non-trivial metric coefficients disappear [22]. These coefficients, which are naturally present in finite-difference methods using non-orthogonal coordinates, are critically large and peaked close to the island separatrices, particularly in elongated island cross sections. A second, global coordinate system defines a 3D mesh with arbitrarily adjustable cell distribution. The mesh is used for scoring Monte Carlo "particles" and for representing plasma parameters. Magnetic coordinates are adopted (if available), as they provide a simple definition of field lines (without need of a time-consuming tracing code) and a magnetic flux consistent distribution of the cells. The described Monte Carlo technique can be viewed as a highly flexible, quite general numerical procedure for modelling strongly anisotropic fluid-transport processes in complex 3D geometries. Recently the EMC3 code has been successully benchmarked with the B2 code.

For high-density plasmas, the EMC3/EIRENE code predicts a detachment of the ionisation front at target temperatures of about 1 eV and a strong toroidal modulation of the plasma parameters (by a factor of ≈ 3) due to the discontinuous targets [23,24]. The prominent role of the cross-field transport for high densities has been confirmed by a detailed transport study with the code, the main results being:

- a drop of the upstream temperature due to cross-field heat transport,
- momentum losses (even at low densities) due to cross-field particle transport,
- a smooth transition from low to high recycling conditions,
- a broad power deposition on the targets (profiles larger than $2\lambda_T/7$ by a factor of 5-6).

Recently, impurity transport has been included selfconsistently into the EMC3/EIRENE code in a first approximation assuming low-Z impurities at small concentrations and a high-density, low-temperature plasma [25]. For an upstream separatrix density of $8x10^{19}$ m^{-3} and a power of P_{sep} = 1 MW entering the scrape-off layer and with the reasonable assumptions of D=0.5 m^2/s, $\chi_e = \chi_i$=3D and a sputtering coefficient of 1.5%, partial detachment was found with 80% power load reduction on the divertor plates by carbon radiation. These upstream plasma conditions have already been achieved experimentally during the divertor pre-investigations with the inboard plates (13).

Additionally to EMC3/EIRENE, a new 3D SOL transport code (BoRiS), based on the same plasma fluid approach as that used for B2/EIRENE, is under development [26]. After the first step, which solves the anisotropic Laplace equations for T_e and T_i , the code has now been extended to include the continuity and momentum equations.

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