Divertor Concept for the WENDELSTEIN 7-X Stellarator: Theoretical Studies of the Boundary and Engineering.

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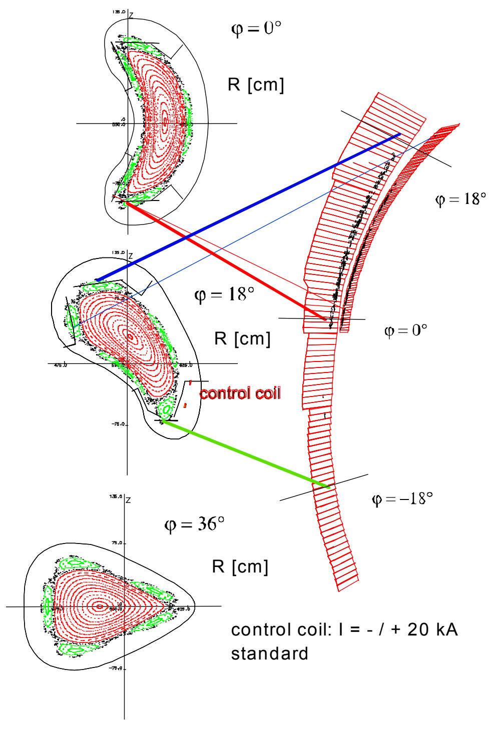
Abstract: The stellarator WENDELSTEIN 7-X is under construction in Greifswald. For energy and particle exhaust under stationary conditions ten divertor units are arranged along the helical edge of the five-fold symmetric plasma column. In respect of the boundary variations in a first step an "open" divertor was chosen. The geometry and the specifications of the in-vessel components reflecting the 3D topology of the boundary are defined in accordance with results of various theoretical studies. The theoretical methods for the characterisation and the proposed technical solutions for the main components - including the instrumentation - target, baffle, wall protection, and control coils will be described.

1. Introduction

W7-X is a large "advanced stellarator" of the HELIAS-type (R=5.5 m, a=0.55 m, B_o=3 T, five periods, moderate shear and variable rotational transform $5/6 \leq \iota \leq 5/4$ at the boundary) [1, 2] with the aim to demonstrate the reactor potential of this stellarator line at steady-state operation close to fusion relevant parameters [3]. Within a wide range of magnetic parameters, using different heating scenarios with possibilities of current drive, gas feed, pellet injection the optimised properties of the HELIAS configuration [4] will be proven. The divertor design will become an important tool to control the plasma parameters during stationary operation.



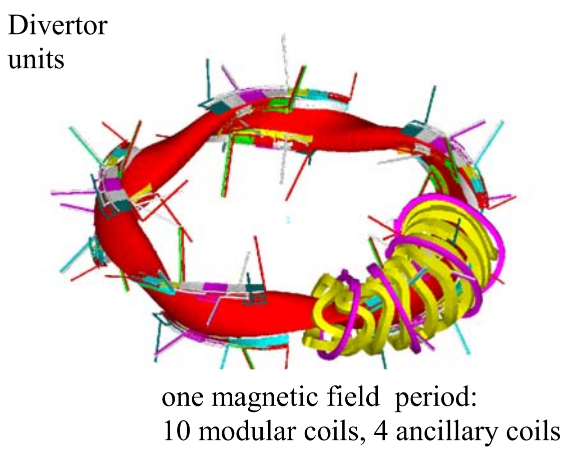


FIG. 1. Divertor and coil system in W7-X. Total view: Plasma contour (LCMS), coils system of one period and 10 divertor units of W7-X (right part). Plasma cross section for different toroidal angles of one half period of the magnetic configuration. For the case ι = 1 =5/5 the deposition pattern on the 3D shaped target plates of one divertor unit is characterised (left part).

2. Divertor concept and modelling

The confinement region in W7-X is either defined by the inner separatrix of islands (being intersected by target plates) or by an ergodised boundary with remnants of islands. Unlike in tokamak divertors, the X-lines in island divertors are helical, with the pitch depending on the resonant rotational transform of the island chain. For the standard configuration (ι = 1 = 5/5 at the boundary) of W7-X, five toroidally closed helical X-lines are present. In the case of extended islands, the positioning of divertor elements along the helical edge (areas with the strongest poloidal curvature of the magnetic surfaces) allows to concentrate the plasma flow from the confinement region in the SOL on target plates and to uncouple the plasma core from the wall completely.
In the first step an **"open divertor" system** [5] for a stationary input power of 10 MW has been studied to be integrated inside the inner cryostat vessel. The optimisation of the 3D divertor geometry is based on the **field line tracing** for the vacuum configurations and the simulation of perpendicular transport by **"field line diffusion"** (Monte-Carlo code) to calculate the power deposition on the targets, whereas the neutral particle balance and the pumping efficiency is modelled using the **EIRENE code** [6].
Finite $<ß>$ plasmas modify the boundary. While the vacuum magnetic field can be easily obtained from the external coils using Biot-Savart's law, a new code, **MFBE** [7], was developed for the computation of finite-β magnetic fields. It evaluates these magnetic fields by using the results of the NEMEC free-boundary finite-β equilibrium code. The sufficiently small shift of the magnetic surfaces and the small change of the rotational transform with increasing β are important properties to guarantee that the target and the baffle plate configurations work without geometrical adjustments for various finite-β equilibria. With increasing plasma pressure and higher rotational transform the **stochastic edge region** of W7-X is widening [8]. In all investigated cases of magnetic parameters and up to a plasma pressure of $<\beta> = 5\%$, the divertor plates intersect the "islands" because the O- and X-points of the islands hardly change their positions in space: The deposition area changes depending on the twist and symmetry of the X-lines, but the defined target areas remain the interacting surfaces for the open peripheral flux bundles controlling energy and particle flows.
The target area are shaped for an incident angle of up to 3° for the magnetic flux hitting the surface. Leading edges are avoided by an appropriate 3D smoothing of the surface contour. Local power densities up to 8 MW/m^2 on the target plates were obtained for a wide range of magnetic parameters, including the worst case at low-density and high-temperature operation. Studies by means of the **B2/EIRENE** code demonstrated a significant unloading of the targets by radiation, especially taking into account low-Z impurities [9].
First attempts to analyse the complex boundary physics in a self-consistent manner including transport on the basis of the 3D plasma transport **EMC3** (Edge Monte Carlo 3D) have been started with promising results [10] for W7-AS, the precedent device of W7-X. The Greifswald stellarator group is progressing with the development of a **3-D plasma fluid model** based on the W7-X topology where strong stochastic effects become important [11].
So, far, the divertor design of W7-X provides a flexible solution of the energy and particle exhaust of a HELIAS device [12]. Unique features to study plasma boundary and transport are included. The variation of the magnetic configuration and the control of the neutral particle balance (by pumping, gas feed) are valuable tools for investigations of the physical phenomena at the boundary to optimise the operation with respect to a reduction of impurity reflux and of power load to the target plates during long pulse discharges. In stellarators

without MHD limitations by disruption - operation at high densities and high radiative losses will be favoured and may allow a significant reduction of the power load to the target plates.

3. Design of plasma facing components

The components, related to the divertor units of W7-X are:

- the **P**lasma **F**acing **C**omponents (PFC): targets, baffles, wall protection
- the pumping system, including cryo panels (Fig. 2)
- control coils.

The "open divertor" system for stationary operation with an input power of 10 MW is designed to be integrated inside the vessel. As plasma facing material graphite for the high heat load components (HHL), B_4C and CFC, low Z material, will be used. Three loading areas with different heat load can be identified. The total plasma facing area is 220 m².

3.1. Energy exhaust

In Fig. 2 an overview about the in-vessel installation is presented: the left side illustrates a cross section of the poloidal plane φ=9°, the right side shows a map of the HHL components on the inner vessel wall for one of the five periods (module #4) related to the poloidal angle theta: –180° to +180° and toroidal angle phi: –36° to +36°, as seen from the magnetic axis

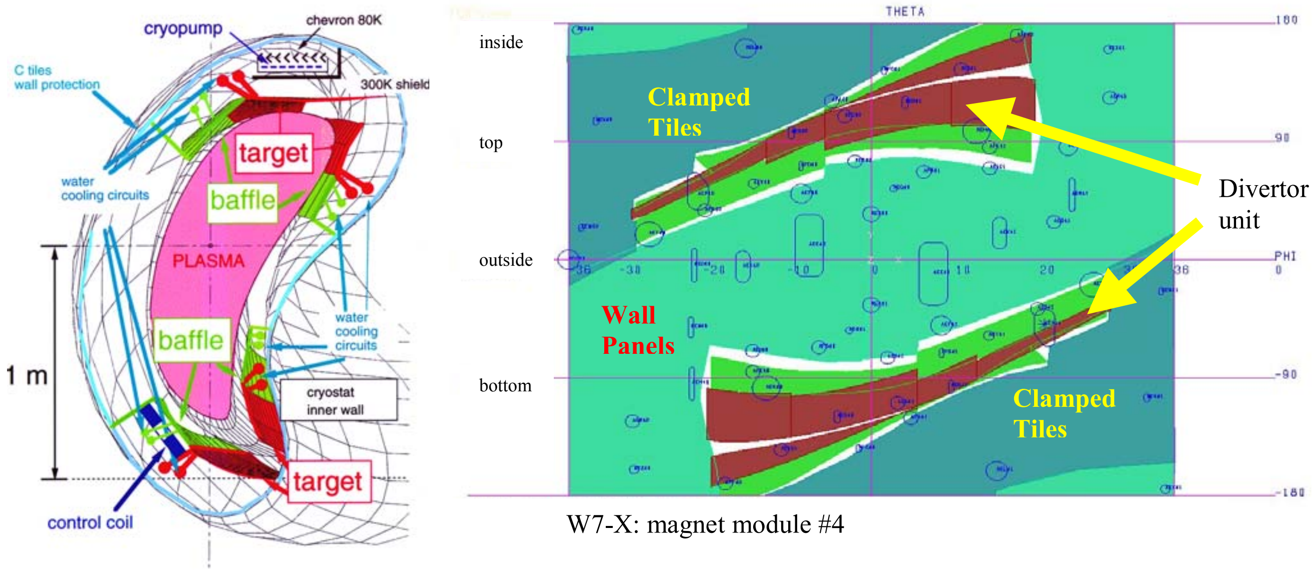


FIG 2: Schematic drawing of the in-vessel components at the toroidal plane φ=9° (left part). Note, the plane is not a symmetry plane. The target area, baffle area and the wall protection is marked. The location of one control coil and one cryo-panel for pumping can be identified.
The right part of the drawing presents the PFC covering the inner vessel in a theta-phi plot of the module 4 in the poloidal and toroidal extension. The positioning of the vessel ports is indicated.

The **target plates** are designed to withstand a heat flux up to 10 MW/m^2 and a maximum total load of 10 MW for steady state operation [13]. Ten divertor units, i.e. two units per period, are necessary in respect of the stellarator symmetry. Each unit consists of two smooth target plates for high power load and baffle plates, designed for lower power loads, adjacent to each plate. The 3D ideal target surface is approximated by flat target elements. The single target elements are standardised and seven different types are foreseen. The averaged width is 55 mm and the length between 270-500 mm. Flat carbon fibre tiles are brazed or welded on the cooling structure. The combination of CFC NB31 and a water cooled CuCrZr heat sink has been selected [14]. Prototypes of target elements, combining CFC with various metallic heat sinks, were already successfully tested for a stationary power load of 12 MW/m^2.

			area	specified load
divertor				
	target plates		30 m²	10 MW/m²
	baffle plates		32 m²	0.5 MW/m²
wall protection			115 m²	
		inboard: graphite tiles	45 m²	250/(500) kW/m²
		outboard: SS panels, B_4C coated	70 m²	200 kW/m²

TABLE 1: SPECIFICATIONS OF THE PLASMA FACING COMPONENTS (SEE FIG. 3)

Baffle plates concentrate the neutral particle flux to improve the pumping efficiency of the divertor [15]. The preliminary concept comprises 32 m² baffle plates. Now, the design is completed in detail and improved for stationary heat loads of 400 kW/m² and local spots up to 500 kW/m². The arrangement of baffle elements follows the modular concept of the target elements. The length of baffle elements ranging from 50 to 300 mm, the width is on average 100 mm. Baffle elements consist of flat fine grain graphite tiles with a maximum length of 200 mm clamped on CuCrZr structures with brazed SS cooling tubes (Fig. 3). Graphite covered gaps of 30 mm width between neighbouring cooling structures allow the integration of plasma diagnostics and the inspection of the divertor chamber after removing the graphite tiles. Fine grain graphites with improved thermo-mechanical behaviour allow the substitution of the formerly proposed CFC. Also the thermo-mechanical behaviour of this fine grain graphite prototypes was successfully tested [16].

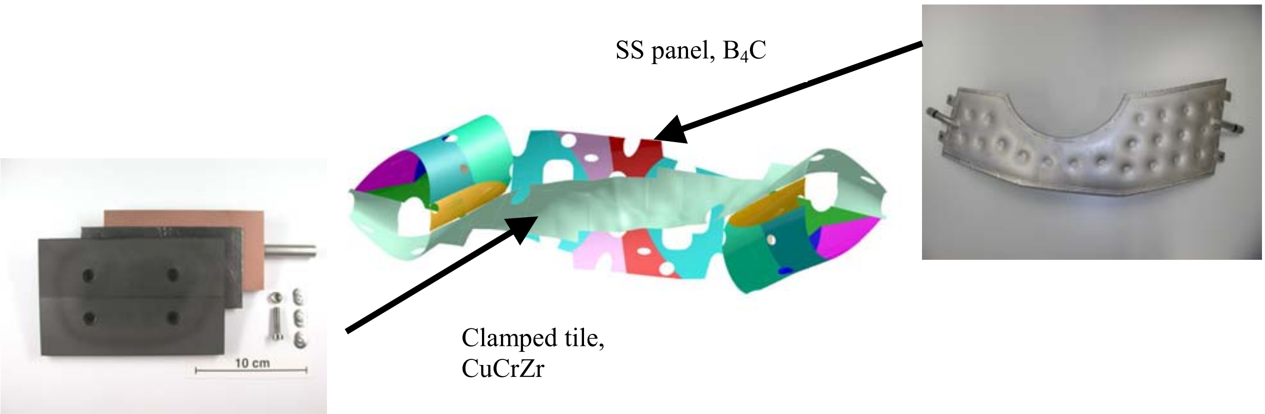


FIG. 3: Wall protection: Inner vessel wall areas protected by the clamped tile elements and panels (cylindrical segments).

In respect of the geometrical situation with locally short distances between the plasma boundary and the cryostat wall and strong curvature of the vessel the baffle concept has to be combined with a **panel concept** to protect the inner wall. Radiation and neutrals at operation with high plasma densities lead to stronger interaction with the wall together with an unloading of the targets. The heat removal capacity of the components is outlined for a total input power of 10 MW with a stationary power load of $P/A < 0.5$ MW/m^2 at the inner narrow regions, $P/A < 0.2$ MW/m^2 for the outer regions where with lower curvature larger elements can be provided. Some prototypes of double wall cylindrical segments have been manufactured and tested (Fig. 3). The shape of the segments with individually defined radius has been optimised to cover the 3D vessel contour. Openings for the ports and additional elements for the port protection are necessary.

3.2. Particle exhaust

The divertor units are completed by the integration of baffles and cryopumps, defined for efficient pumping under stationary conditions dealing with external fluxes of 5 10^{21} 1/s (NBI, pellet, gas feed etc.) and pressures in the range of up to 10^{-3} mbar in the divertor units [17].
For safe operation of the divertor components operational diagnostics, like thermography, thermometry, water flow control, measurements of thermo-currents, gas pressure and mass spectrometers will be installed. Supplementary sophisticated diagnostics will be used to explore the plasma parameters of the boundary and at the interacting areas.

4. Conclusions

The stellarator W7-X is designed for steady-state operation. Within a wide range of magnetic parameters and applying different heating scenarios with possibilities of current drive, gas feed, pellet injection the optimised properties of the HELIAS configuration will be proven. The adaptation for a wide operational range of the magnetic parameters requests an "open divertor". The properties of the magnetic configuration allows to select a divertor geometry without needs of adjustment dependent of the particular plasma parameters and <ß>. Additionally, experimental flexibility is provided by means of control coils.
The divertor design of W7-X offers a flexible solution of the energy and particle exhaust of a HELIAS device. The construction of the divertor components is feasible. The variation of the magnetic configuration and the control of the neutral particle balance (by pumping, gas feed, impurity doping) are valuable tools for investigations of the physical phenomena at the boundary to optimise the operation with respect of reduced impurity reflux and power loads to the target plates during long pulse discharges and high input power. In stellarators without MHD limitations by disruption - operation at high densities and high radiative losses will be favoured and may allow a significant reduction of the power load to the target plates.
The further optimisation of the divertor will be a major goal of the activities in W7-X. A „closed divertor“ may be tested in a second step in W7-X and could become a solution for a HELIAS reactor [18].

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