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Wendelstein 7-X—Status of the project and commissioning planning

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ABSTRACT

The stellarator device Wendelstein 7-X (W7-X) is now in the final stage of assembly. All the machine components have been built and in mid 2014 the commissioning activities should start. The first objective of W7-X is to prove the stellarator optimization principles, i.e., to reach at least the same confinement quality as a similar sized tokamak. The second objective is to demonstrate stable high-power steady-state operation. In the present paper, after a short description of the device, the ongoing activities to complete the construction will be summarized, focusing on the optimization of the layout of the torus hall and on the completion and assembly of the in-vessel components. The main lessons learned during the assembly will be presented as well as the application of the quality control, the handling of non-conformities and the design change procedures. The planning of the commissioning sequences to perform the integral tests of the major W7-X systems is discussed in order to prepare the W7-X device to start plasma operation.

1. Introduction

The Wendelstein 7-X (W7-X) fusion experiment is the next step device in the stellarator line of Max-Planck Institute for Plasma Physics and is presently under the final stage of assembly at the Greifswald site. W7-X will be the largest optimized stellarator in the world with a plasma volume of 30 m^3 operating a reactor-relevant plasma under steady-state condition [1,2].

W7-X has been designed with the aim to show the viability of the optimized stellarator concept for future power plants. The first objective is to prove the optimization principles, i.e., to reach at least the same plasma confinement quality as a similar sized tokamak. The second objective is to demonstrate stable high-power steady-state operation. W7-X will also address the critical issue of plasma–wall interaction in long plasma pulses.

The main challenges in the construction of W7-X derive from the complexity of the 3D geometries, high dimensional accuracy required, the limited available spaces and the high loads in the magnet system [3].

The main parameters of W7-X are shown in Table 1.

The construction of the machine is almost completed [4–6]. A schematic diagram of W7-X is shown in Fig. 1.

In order to start the commissioning of the machine and the plasma operation as soon as possible, it has been decided to start with a simplified configuration in the so-called Operational Phase 1 (OP1), which is divided in two phases: OP1.1 and OP1.2. The objective of OP1.1 is to measure the precision of the magnetic field and to test all the systems of the machine including periphery components, some basic diagnostics, the Electron Cyclotron Resonance Heating (ECRH) and the Control Data acquisition and Communication (CoDaC) system. The main goal of OP1.2 is to develop credible and stable discharge scenarios to demonstrate the feasibility to reach the stellarator optimized parameters (confinement and stability properties, divertor loads, high density discharges). The simplified configuration adopted in OP1.2 mainly consists in:

- installing radiation cooled Test Divertor Units (TDUs) allowing for 5–10 s plasma pulses;
- installing heating systems with up to 8 MW of ECRH and 3.5 MW of Neutral Beam (NB);
- installing a first set of diagnostics.

For the second operational phase (OP2) more heating power will be available and the TDUs will be replaced by the steady-state capable, actively cooled High Heat Flux (HHF) divertor able to withstand up to 10 MW/m^2. OP2 should demonstrate the high-power steady-state operation of W7-X and therefore foresees also ten cryo-pumps in the divertor.

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Table 1
Key engineering parameters of the stellarator W7-X.

Engineering parameter	Parameter value
Major radius	5.5 m
Minor radius	0.53 m
Number of non-planar coils	50
Number of planar coils	20
Number of current leads	14
Number of ports	254
Machine height	4.5 m
Outer diameter	16 m
Total mass	750 t
Total cold mass	425 t

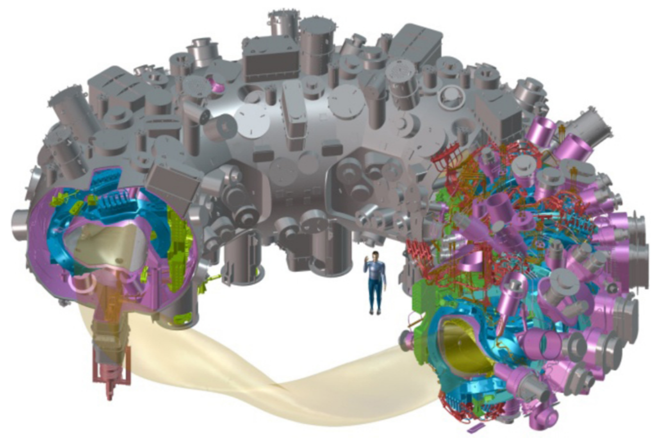

Fig. 1. Schematic diagram of W7-X.

2. The W7-X machine

W7-X is a superconducting fusion device with a five-fold magnetic field periodicity, which corresponds to five nearly identical modules, each consisting of two flip-symmetric half-modules with five different Non-Planar Coils (NPCs) and two different Planar Coils (PCs) each. The coil support structure of W7-X [7] is rather complex because it is the result of a compromise which guarantees the integrity of the magnetic system under limited deformations with acceptable stresses. Each PC and NPC is supported by the central support structure that has to keep the coils at their precise position. The Plasma Vessel (PV) has a shape that closely follows the twisted shape of the plasma cross section which varies between triangular and bean shaped. 254 ports are connecting the PV to the Outer Vessel (OV) and are divided in two types: diagnostic ports which also include plasma heating ports and supply ports. The thermal insulation, made by multilayer insulation supported by actively cooled thermal shields kept at 60 K by helium gas, is located around the PV, the ports and the OV. The plasma-facing components include the divertor made by target plates and baffle plates for an area of 19 and 33 m^2 respectively and the wall protection made by graphite tiles (inboard) and stainless steel panels for a total surface area of 115 m^2.

The basic coil system of W7-X [8] consists of 50 NPCs, 20 PCs and a bus-bar system connecting the coils of each type in series to their respective power supply through current leads. To energize the NPCs and PCs seven power supplies, independently adjustable between 0 and 20 kA, are used in order to achieve the required magnetic configuration. For each of the ten divertor modules a control coil is foreseen for strike-point position control. The control coils are made by water-cooled copper hallow conductor and are separately energized. Being located close to the plasma surface, the control coils can correct mainly the Fourier components B_{33} and B_{44} of the error field [9]. In addition five normal conducting so-called trim coils are installed outside the cryostat and are independently supplied in order to adjust the Fourier component B_{11} and B_{22}. Both sets of coils are used to correct asymmetries due to the tolerances of fabrication and assembly of the superconducting coil system. The PV is composed of 10 half-modules. Each half-module is split into two sectors to allow the assembly of the first NPC and it is made by 20 stainless steel (SS) segments precisely bent and welded one to the other. The 254 ports have different forms; round, oval and rectangular; each one is equipped with a bellow to allow for thermal expansion during baking. Water pipe loops welded on the SS walls of the PV and ports are used for the baking at 150 °C and for the cooling during plasma operation. The PV is supported vertically by 15 legs which allow it to move radially during baking and to be adjusted vertically with respect to the coil structure by ±5 mm.

The In-Vessel Components (IVCs) [10,11] include the plasma-facing components (divertor targets, baffle modules, heat shields, panel wall protection and special heat shields in the Neutral Beam shine-through areas), a number of embedded diagnostics (Rogowski coils, Mirnov coils, diamagnetic loops), the control coils and the cryo-vacuum pumps. The divertor system consists of 10 similar units composed of nearly horizontal and vertical target modules and baffle modules adjacent to the target. The plasma vessel wall is protected by heat shields and water cooled stainless steel panels. The latter are located in areas well away from the plasma and are designed to remove an average heat flux of 100 kW/m^2. The low heat flux central part of the divertor (1000 kW/m^2), baffle (500 kW/m^2), and heat shields (300 kW/m^2) are built using the same basic technology: fine grained graphite tiles are mechanically clamped onto CuCrZr alloy heat sinks using TZM screws. The HHF divertor target elements are designed for a peak heat flux of 10 MW/m^2. The HHF divertor consists of 100 HHF target modules, 10 per divertor unit. Each module consists of a set of 8–12 target elements assembled onto a support frame and fed in parallel from a water manifold.

3. Assembly of the machine

3.1. *Plasma vessel, outer vessel, ports*

The PV consists of 20 sections. That enables the threading of the NPCs. Every coil requires a complex threading path (tilting, rotation, shifting) in all 6 degree of freedom. The minimal acceptable (theoretical) distance between coil and PV segment was set to 4 mm. Every PV section is insulated before it is moved to the assembly area. PV sections have to be connected to each other with an accuracy of about 2–3 mm. These sections are not very rigid because of the many openings for the later port installation. Dead weight deformations of the PV sections of up to 3 mm were observed. To compensate for it complex stiffeners had to be developed and optimized.

The first two sections are welded together once the first coil is threaded. Two sections represent a so-called half-module. The two PV sections have been aligned to each other and stiffened temporarily. The weld is made from inside and outside (X-prep, 17 mm wall thickness). The contour of the PV sections is surveyed by Laser Trackers (LT), checked against CAD requirements and the sequence of weld layers is adapted accordingly. Partial X-raying, helium leak test with local test chambers and the closure of the thermal insulation complete the work at this connection area. Six further coils are threaded across the two welded PV sections [12] (see Fig. 2). Thereafter all 7 coils are mechanically connected with each other forming one half-module of the magnet system.

Two half-modules are connected together forming a module of W7-X. The two PV half-modules are aligned to each other by means of a system of rods and bearing. The gap between both vessel parts