Optimized Stellarator as a Candidate for a Fusion Power Plant

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1. Fusion

The aim of fusion research is to develop a new primary energy source, which has great potential to significantly contribute to the global energy supply. The sun, a "regular" star of the G2-type, is mainly powered by the fusion of protons to Helium. The extremely small reaction rates of this process require very high particle densities. Additionally, in order to overcome the Coulomb barrier between the charged particles by means of the tunnel effect, core temperatures of about 1 keV are required. This results in plasma pressures of the order of 1011 bar that are impossible to handle on earth under steady-state conditions. The fusion reaction of Deuterium (heavy isotope of hydrogen) with Tritium (super-heavy isotope of hydrogen) has a much more favorable reaction cross-section at reasonable particle energies (about 10 keV). The reaction scheme reads to be

$$_1D^2 + {_1}T^3 \rightarrow {_2}He^4 \ (3.5 \ \text{MeV}) + {_0}n^1 \ (14.1 \ \text{MeV}).$$

This reaction generates about 17.6 MeV of kinetic energy per reaction. Radioactive isotopes (in particular lanthanides and actinides) are not produced. The Helium (α) particle provides the heating of the plasma to maintain the fusion conditions. The neutrons are carrying most of the generated energy and will be used in a power plant not only to transform kinetic into thermal energy but also to produce Tritium via the following nuclear reactions with Lithium:

$$_3Li^6 + {_0}n^1 \rightarrow {_1}T^3 + {_2}He^4$$
$$_3Li^7 + {_0}n^1 \rightarrow {_1}T^3 + {_2}He^4 + {_0}n^1$$
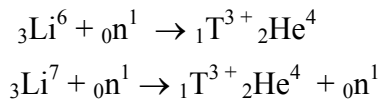
The process of breeding tritium happens in the so called "breeding blanket" – an about 2m thick layer on the wall of the reactor chamber. In the blanket, the neutrons are slowed down, the kinetic energy is absorbed and a cooling fluid (or gas) is heated up which finally – after passing a heat exchanger – drives turbines to produce electricity. The blanket also contains the Lithium to react with the neutrons for producing Tritium. The Tritium will be separated from Helium and fed back into the plasma. It is important to note that the radioactive Tritium (β-radiator with 12.3y half value time) is produced in-situ and needs not to be transported to the power plant.

The ultimate fuel for a future fusion power plant is Deuterium and Lithium. Deuterium and Lithium are available throughout the world in almost inexhaustible quantities (in the sea water and in the earth) and can satisfy the world's energy demand for millions of years.

To ignite the fusion reaction, appropriate conditions are required: One has to achieve high plasma temperatures (10 to 20 keV), sufficiently high particle densities (about 10^{20}m^{-3}), and good energy confinement times (several 10 s) simultaneously. The energy confinement time is a measure for the heat insulation between the plasma and the (cold) wall. These conditions can be reached using magnetic confinement of the plasma (see below).

2. Magnetic confinement

Plasmas can be shaped by magnetic fields because they consist of charged particles: the ions and the electrons both follow helical trajectories around the magnetic field lines due to the Lorenz force. The particles are thus tied to the field lines but move force-free in the longitudinal direction of the magnetic field lines. For example, a long cylindrical coil produces straight magnetic field lines; ions and electrons are magnetized and radial losses are strongly reduced. The free motion along the field lines, however, leads to strong particle and energy losses at both ends of the cylinder. By arranging magnetic coils to form a closed ring, a torus-shaped magnetic field with closed field lines is created which avoids end-losses. However, a purely toroidal magnetic field is well-known not to confine a plasma due to strong radial particle drifts. The field lines have to be "twisted" to compensate the charge imbalances that prevent magnetic confinement of the plasma. These twisted field lines form nested magnetic surfaces, on which the density and temperature are constant. Fusion research currently concentrates on devices of two concepts of magnetic confinement, the tokamak (Fig. 1) and the stellarator (Fig. 2).
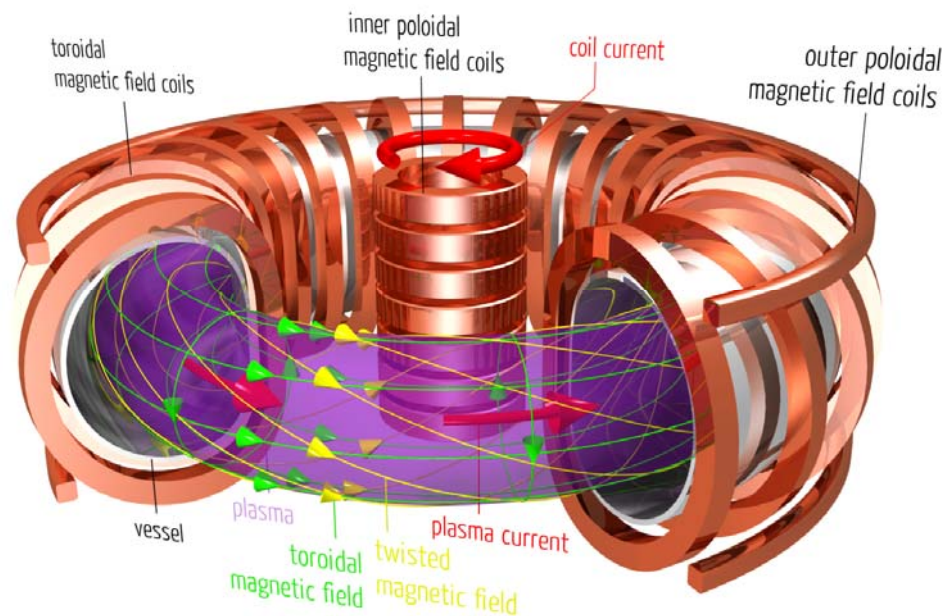

Fig. 1: Schematic drawing of a tokamak. The copper colored rings are magnetic field coils. The plasma is indicated in magenta. Green arrows represent the purely toroidal magnetic field, yellow arrows the twisted magnet field lines. The red arrow represents the strong plasma current induced by the current in the central solenoid.

In a tokamak, the toroidal magnetic field is provided by planar coils. The toroidal field is twisted by the poloidal field generated via an electric current in the plasma. The plasma current is induced by the field generated in a central solenoid that is driven with an alternating current. One could say that the plasma acts as the secondary winding of a transformer. The induced electric field is also used for plasma build-up and heating and it is important to note that a tokamak plasma equilibrium is a self-organized state. Tokamaks have a relatively simple geometry and they are toroidally symmetric, which yields a number of conserved quantities. The tokamak concept has proven to be very successful: today's most advanced experiments are based on this principle. However, the ultimate requirement

to run a strong current in the plasma is a serious obstacle for stable steady-state operation. Since induction requires pulsed operation, the plasma current must be driven by other means, e.g. particle beams or plasma waves. In addition – from the thermodynamic point of view – a strong plasma current is a source of free energy, which is often released via plasma instabilities that can lead to a total breakdown of the plasma equilibrium (current disruption).
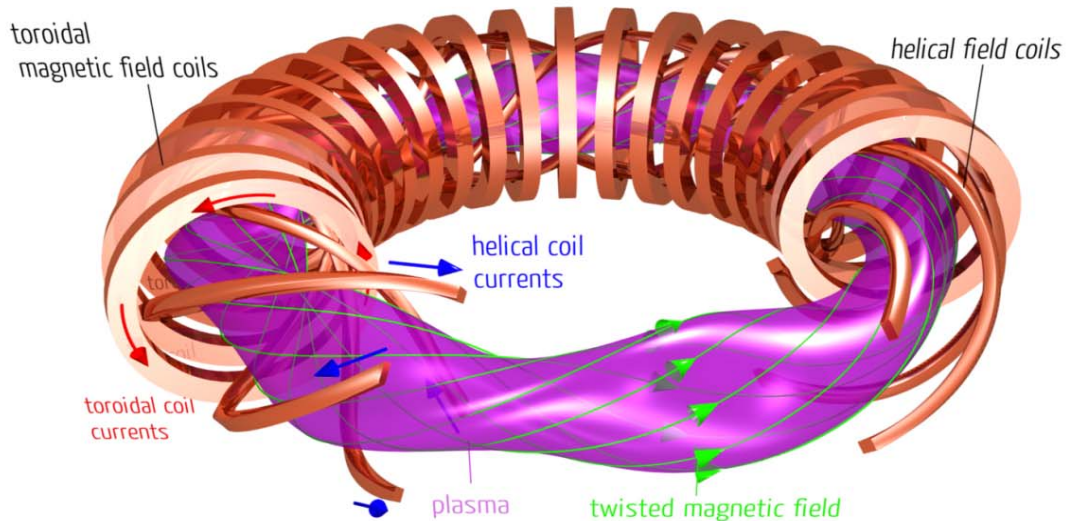

Fig. 2: Schematic drawing of a classical stellarator. The copper colored rings are magnetic field coils. The plasma is indicated in magenta. Green arrows represent the helically twisted magnetic field lines. The red and the blue arrows represents the current in the toroidal and helical coils, respectively.

3. Stellarators

The stellarator is an alternative concept. The stellarator magnetic field is generated by external coils only. A plasma current is not needed. As a consequence, different from a tokamak, the vacuum magnetic field has already confinement properties and the stellarator is intrinsically steady-state capable. The classical stellarator (Fig. 2) is based on a combination of planar coils and a pair of large, helical coils. These helical coils are critical from the engineering point of view (manufacturing, assembly, repair).

Helical coils can be avoided by combining helical and planar coils to non-planar ones, the so-called Wobig-Reker coils. An important benefit of non-planar coils is that the magnetic field geometry can be shaped via the specific coil geometry. This allows for a physics-based optimization of the magnetic field, which turned out to be extremely beneficial for the development of the stellarator line (see below). The optimization of the magnetic field is, however, computationally expensive and only after the first supercomputers of the late 80's became available, the suite of coupled codes could be run in a reasonable time frame. This also explains why the stellarator concept is somewhat lagging behind the tokamak concept, which is mathematically more accessible. Meanwhile, the stellarator research is moving from mid-size to large-scale experiments.

There are seven criteria that were chosen to form the basis stellarator optimization:

1. high quality of vacuum magnetic surfaces,
2. good finite equilibrium properties at $\langle\beta\rangle = 5\%$,
3. good MHD stability properties at $\langle\beta\rangle = 5\%$,
4. reduced neoclassical transport in $1/\nu$ -regime,
5. small bootstrap current in the long-mean-free-path regime,
6. good collisionless fast particle confinement,
7. good modular coil feasibility.

Here, $\langle\beta\rangle$ is the magnetic pressure $p_{mag} = B^2/2\mu_0$ normalized by the plasma pressure $p = n\,k_B\,T$ averaged over the total plasma volume. The $1/\nu$-regime is a parameter range, where the diffusion coefficient scales inversely with the collision frequency ν.
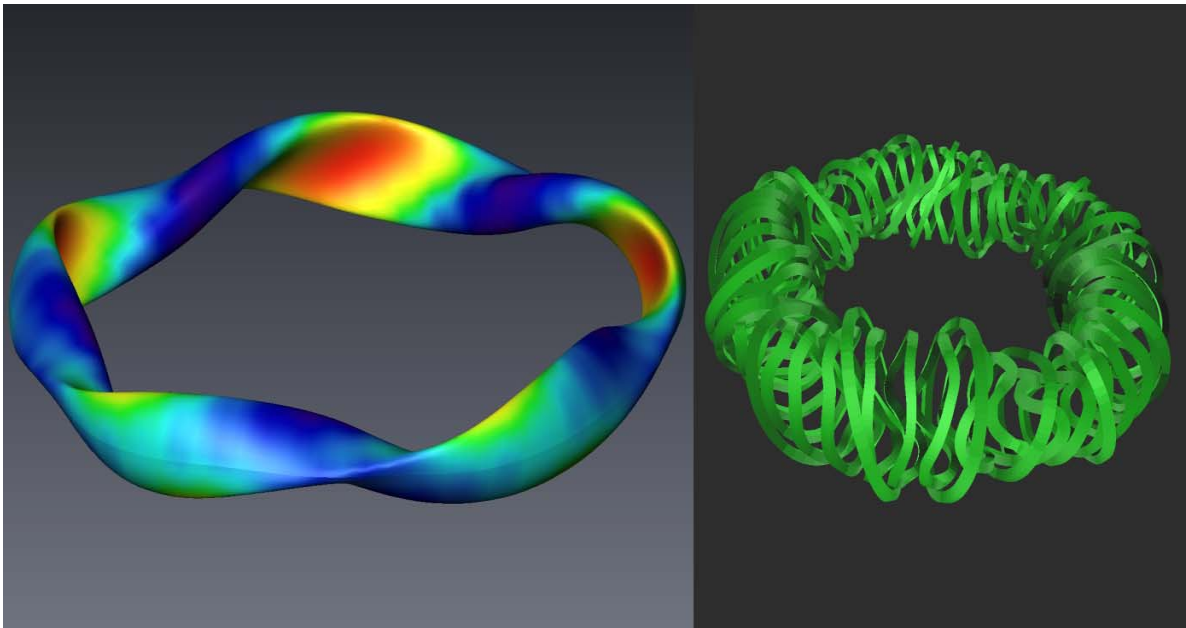
Fig. 3: Color-coded modulus of the magnetic field at the last closed flux surface (left). Red color indicates the strongest field, blue color the weakest field. The corresponding magnet system consists of 50 non-planar and 20 planar coils (right).

Using the above optimization criteria, the magnetic field and plasma geometry of the stellarator device Wendelstein 7-X has been obtained. Fig. 3 shows both the modulus of the magnetic field at the last closed flux surface and the magnetic field coil system with 50 non-planar coils. The coils have five different coil geometries and they are arranged in a five-fold symmetry. Each coil type is separately controlled, which yields a range of magnetic configurations. An additional toroidal field is generated by a set of 20 planar coils, which adds further to the flexibility of the device. It is a major scientific goal of the experiment Wendelstein 7-X to explore the whole range of optimized magnetic geometries with regard to plasma performance and consistency with divertor operation (see below).

4. Engineering

The engineering design of Wendelstein 7-X is grouping the coils into five identical magnet modules. Each module (Fig. 4, 5) consists of two flip-symmetric half modules with five

different non-planar and two different planar coils each. Coils of each type are connected in series via superconducting bus bars and can be energized independently to provide the above mentioned wide range of operational flexibility. Each of the seven coil groups is connected to a power supply via a pair of current leads. The current leads form the interface between the warm end (power supply) and the cold end (bus bar).

The 50 non-planar and 20 planar coils are superconducting (Liquid He cooled NbTi cable-in-conduit with 18.2 kA and 16 kA nominal current, respectively). Each coil bolted to a central support structure with two extensions, which have to take up considerable mechanical stresses of the order 100 MPa. The structural integrity of the complete magnet system requires a variety of different mechanical support elements between each coil pair.
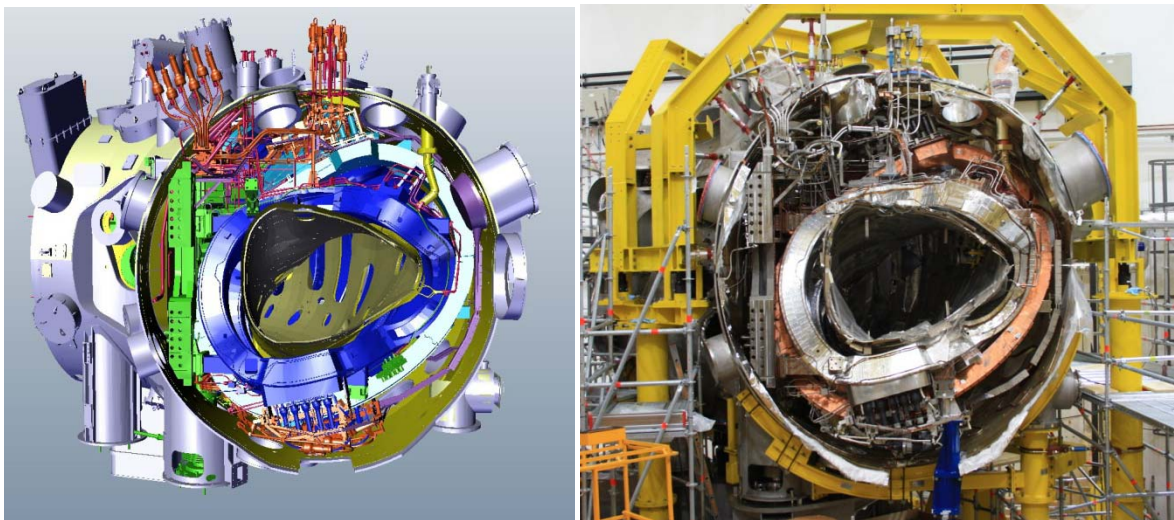
Fig 4: A complete module of Wendelstein 7-X as a CAD model (left) and as built (right).
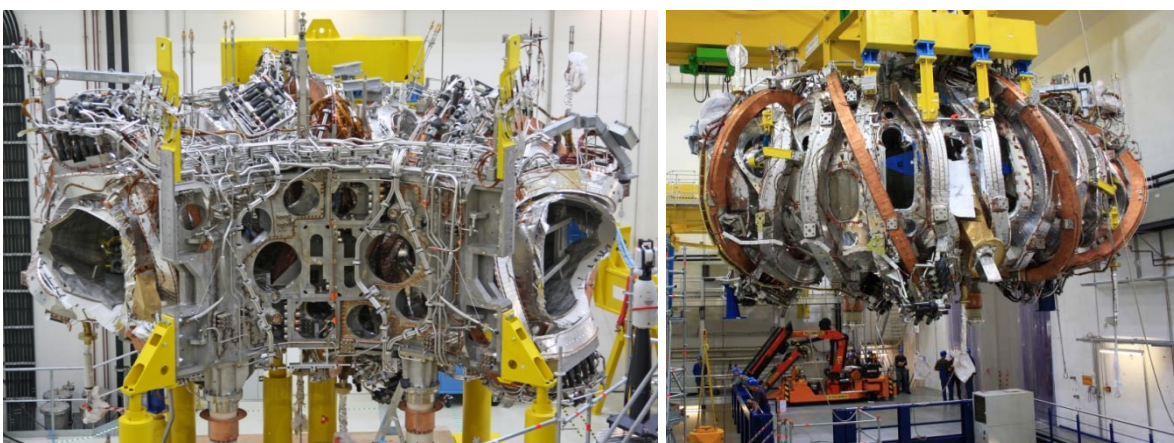
Fig. 5: Magnet module with a view on the central support ring element (left) and hanging on the lifting unit with a view on the non-planar and planar coils (right).

The cryostat provides the thermal insulation of the cold magnet system and consists of the plasma vessel with 30m^3 volume, the cryostat vessel with an outer diameter of 16 m, the 254 ports, and the thermal insulation. The cold mass is 425 t in total and is cooled with liquid Helium to 3.4 K. The ports have over 100 different shapes and dimensions to allow the optimum access from the outside to the plasma vessel. They are used for plasma diagnostics and heating, supply and exhaust. The outer surface of the plasma vessel, the inner surface of the cryostat vessel and all port outer surfaces are insulated with a multilayer insulation of aluminized Kapton® foils and an actively cooled 70 K thermal shield. The thermal shield is made of brass except for the plasma vessel, where a glass fiber composite with copper meshes is used. The purpose of the thermal insulation is to minimize the thermal load on the cryogenic components in the cryostat.

The key element for the controlled contact between the plasma and the solid wall is the so called divertor. A divertor is used to control the power and particle flux from the plasma edge. Wendelstein 7-X is using the island divertor concept, where the natural islands in the magnetic field structure intersect the most highly loaded divertor elements (see Fig.6). Like the magnetic field, the divertor has a five-fold symmetry and is composed of 10 units, one in the top and one in the bottom of each magnetic field period.
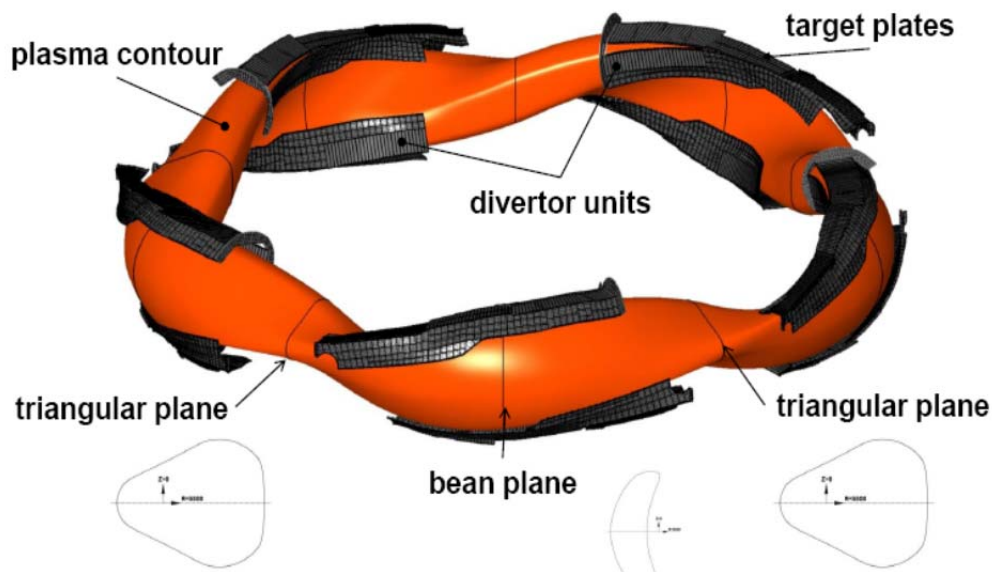

Fig. 6: Schematic drawing of the island divertor of Wendelstein 7-X. It consists of ten divertor modules (black structures: divertor target plates) located at the bean-shaped cross section of the plasma, such that natural magnetic islands intersect with the target plates.

In addition, the steel wall of the plasma vessel is completely covered with actively cooled wall elements: graphite heat shields, steel wall panels, baffle modules – in total 265 m^2 surface. The water cooling is provided by a complex system of water lines with 4000 m length in total. The heat load on the in-vessel components varies between 100 kW/m^2 and 10 MW/m^2 and the total energy turnaround is 36 GJ. Cryopumps are used to remove the neutral gas in the divertor volume and control coils allow one to limit the thermal flux on the most loaded areas of the divertor.

After more than 10 years of construction time, all major device components are manufactured, tested and delivered. The five modules are installed on the machine base and connected with each other (Fig. 7). The remaining large assembly work packages are the following:

- assembly of altogether 2500 large in-vessel components with about 710.000 single parts in total;
- assembly of the 14 current leads;
- assembly of the device periphery, including steel support structures, platforms, water and He pipework, device instrumentation, cable trays, cabels, electrical cabinets, diagnostics, heating systems.

Since the agreement on a sound baseline plan in 2007, the Wendelstein 7-X project has been on budget and on schedule. The completion of the device is foreseen for mid-2014 and the detailed planning for the device commissioning is in progress. The first plasma is scheduled for mid-2015.
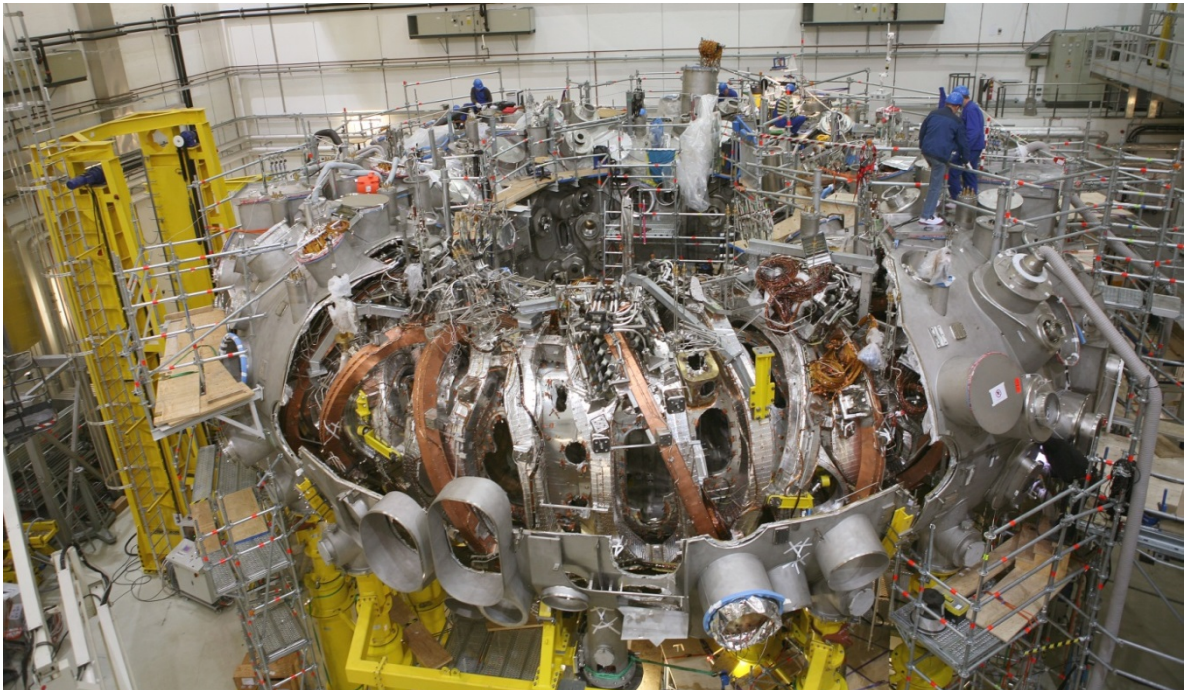
Fig. 7: View on Wendelstein 7-X with all five magnet modules on the machine base. The last half-shell of the cryostat vessel is closed in the mean time.

5. Program

The major scientific goals for Wendelstein 7-X are the following:

- to demonstrate with the construction of Wendelstein 7-X that it is possible to build an optimized stellarator with modular, non-planar coils;
- to prove that a stellarator can confine a plasma as good as a tokamak of similar size;

- to integrate high temperature, high density, good confinement discharge scenarios with good stability, fully consistent with divertor operation, fuelling, density control and particle exhaust;
- to maintain fusion relevant plasma parameters for about thirty minutes and thus to prove that an optimized stellarator can be operated under high-power steady-state conditions.

This is an ambitious program that intends to position the optimized stellarator as a candidate for a future fusion power plant. However, Wendelstein 7-X cannot address all scientific and technical questions alone. In particular, it is not a nuclear device, i.e., Deuterium-Tritium (D-T) operation is not foreseen and technically not possible. For addressing the pressing questions of integrated nuclear operation with a D-T plasma, one needs a much bigger device in order to obtain a relevant level of α-particle heating. This is one of the main goals of the large, international project ITER, under construction in Cadarache, France (Fig. 8).

The tokamak ITER (latin for "the way") is to generate 500 megawatts of fusion power, which means about a factor of 10 more energy than necessary for heating fusion plasma. ITER is currently being built by the European Union, Japan, the USA, China, Russia, India and South Korea. Its construction is estimated to take about ten years.
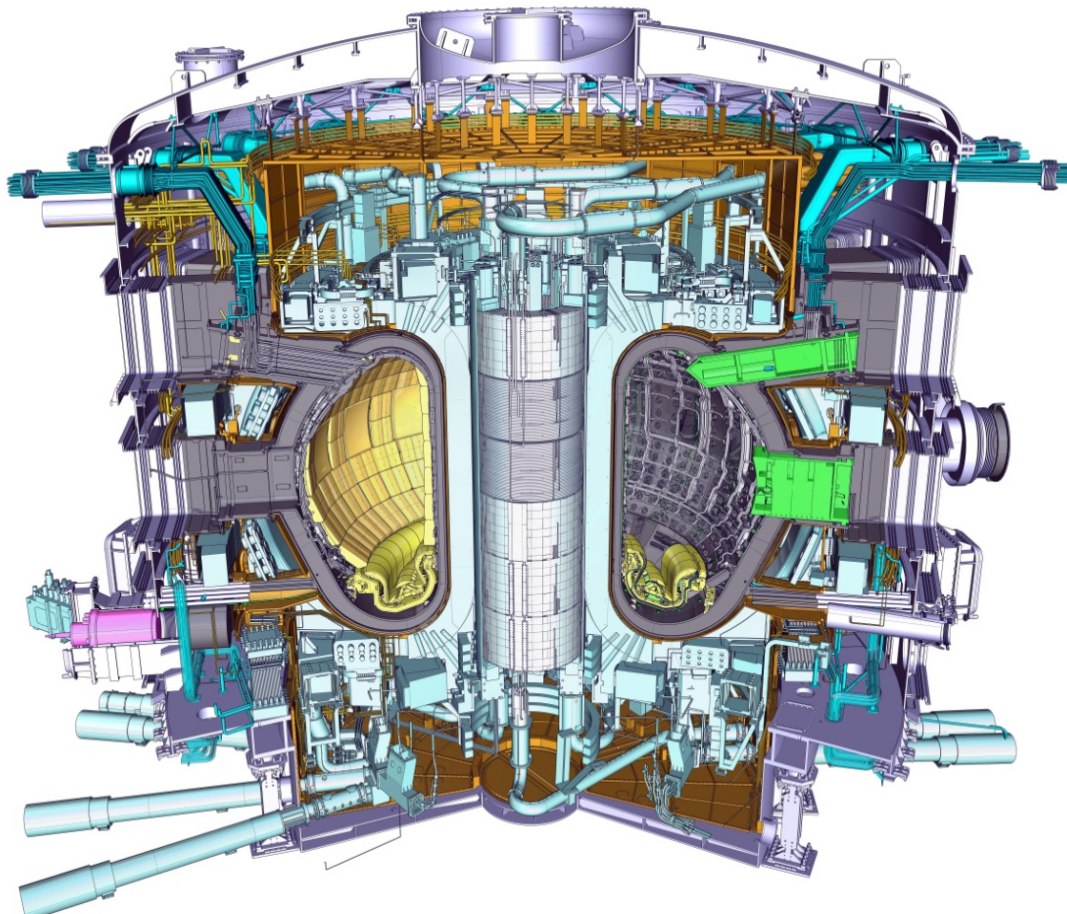

Fig. 8: ITER – the world's largest tokamak with a plasma volume of 840 m^3 (courtesy: ITER.org)

An extrapolation of Wendelstein 7-X to a stellarator fusion power requires ITER results, both in physics (α-heating) and technology (Tritium breeding). A stellarator fusion power plant based on the present day knowledge gives an idea what such a device could look like (Fig. 9 and Table 1).

	Wendelstein 7-X	Stellarator FPP	Unit
Fusion power	0	3000	MW
Toroidal magnetic field	3	5-6	Tesla
Plasma volume	30	1500	m^3
Heating power	20-30	600 (α-power)	MW
Average neutron flux	0	1	MW/m^2
Average heat flux to invessel components	0.1	0.4	MW/m^2

Table 1: Key parameters of this stellerator fusion power plant (FPP) in comparison to Wendelstein 7-X
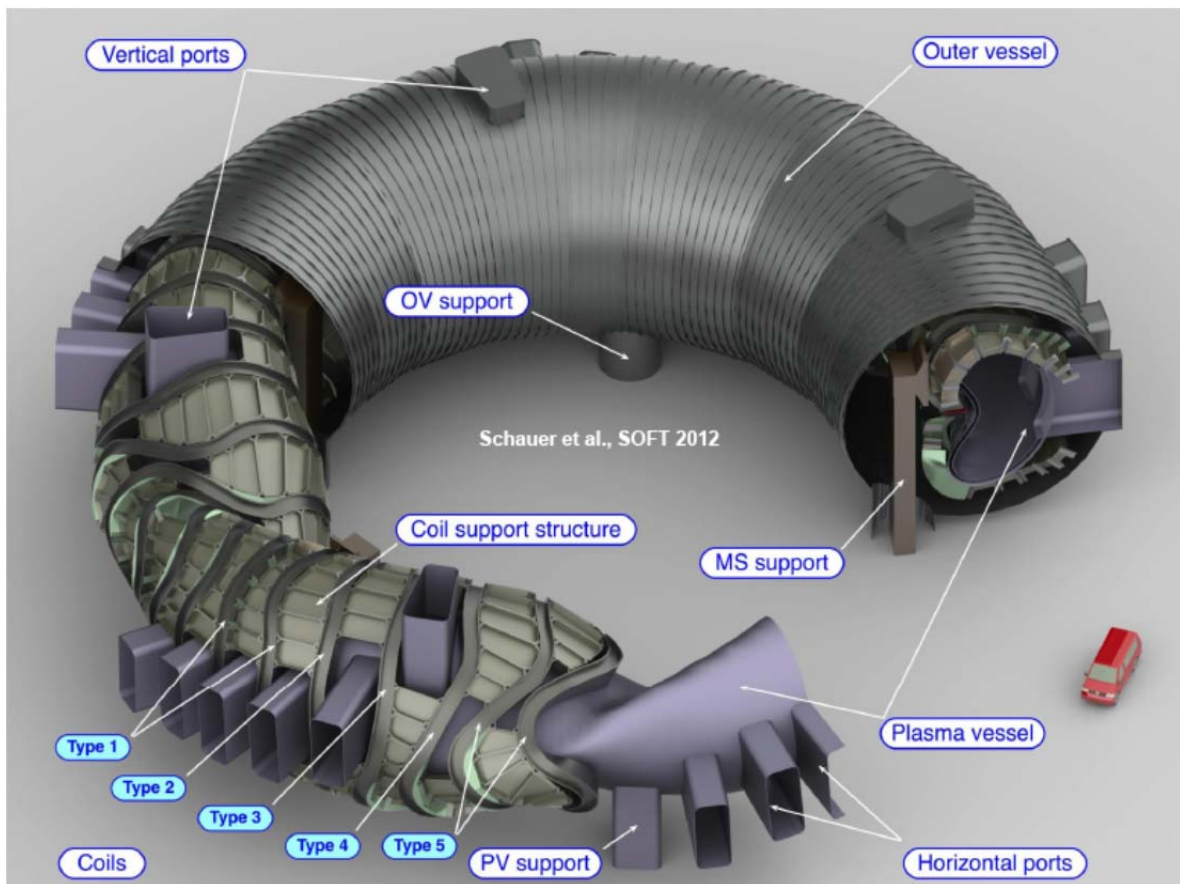
Fig. 9: An artist's view of a stellarator fusion power plant with about $1500m^3$ plasma volume and an outer diameter of about 45m.

6. Conclusions

Wendelstein 7-X is a key device for fusion in general and for the stellarator concept in particular. The start of commissioning is foreseen for spring 2014 and the scientific operation will start in spring 2015. It is the goal of the project Wendelstein 7-X to demonstrate the reactor potential of the stellarator concept. For that purpose, the careful optimization of the magnetic field geometry is indispensible. Wendelstein 7-X will be the first experimental fusion device that is able to operate fusion-relevant plasmas under steady-state conditions.

Further reading

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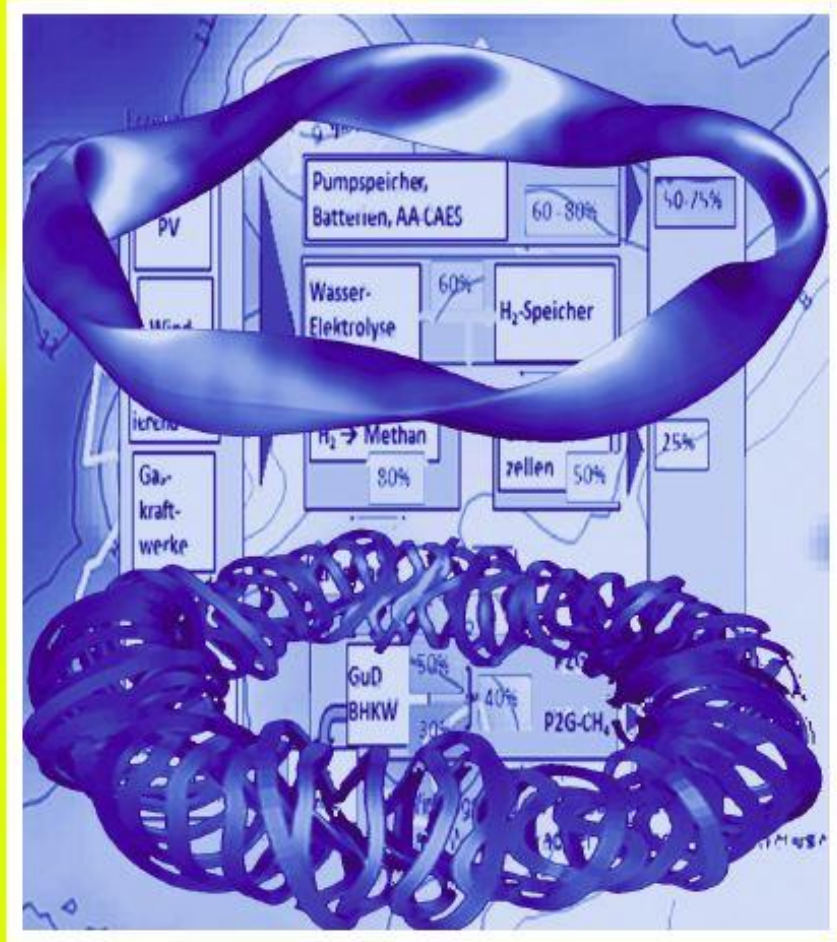
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Haupt- und Plenarvorträge

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