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Challenges on the road towards fusion electricity

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The ultimate aim of fusion research is to generate electricity by fusing light atoms into heavier ones, thereby converting mass into energy. The most efficient fusion reaction is based on merging the hydrogenic isotopes: Deuterium (^{2}D) and Tritium (^{3}T) into Helium and a neutron, which releases 17.6 MeV in the form of kinetic energy of the reaction products (see Fig. 1).

The two main strategies to achieve fusion on Earth are based on magnetic confinement and inertial confinement. In *magnetic* confinement, a gas is heated to temperatures in the order of 1 - 1.5 × 10^{8} K. At these high temperatures the gas has transformed into plasma, consisting of charged particles with sufficiently high energy to fuse. Magnetic fields are used to confine the plasma and keep it away from any material surfaces. In *inertial* fusion a small pellet of solid deuterium-tritium is quickly and strongly compressed by powerful laser or particle beams, leading to sufficiently high densities and temperatures for fusion.

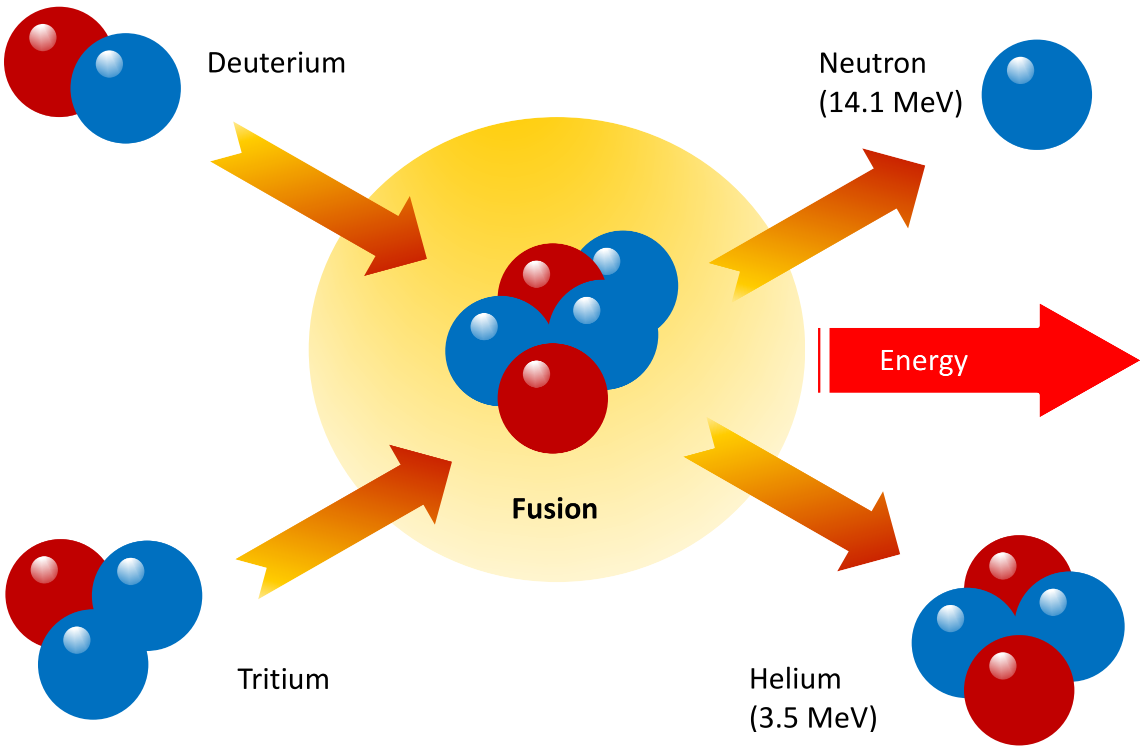

Fig. 1: Fusion reaction between deuterium and tritium. The helium particle carries 20% of the reaction energy which is used for heating the plasma. The neutron with 80% of the energy is not confined by the magnetic field and will penetrate into the blanket surrounding the plasma. There it deposits its energy, leading to a temperature rise of the blanket coolant, which will drive electric turbines. In the blanket it also converts ^{6}Li into ^{3}T and ^{4}He; the ^{3}T is subsequently used as fuel.

Magnetic Confinement Fusion

European fusion research is largely concentrated on magnetic confinement fusion, as it is the most promising concept to deliver fusion electricity. In the range of magnetic confinement devices that have been studied over the last decades, the tokamak has reached the best performance [1,2]. In a tokamak, the plasma is confined by a magnetic field that is a superposition of a field generated by external magnetic coils (yielding a field in the toroidal direction) and an internal poloidal field generated by a toroidal current through the plasma which is induced by a transformer (see Figure 2).

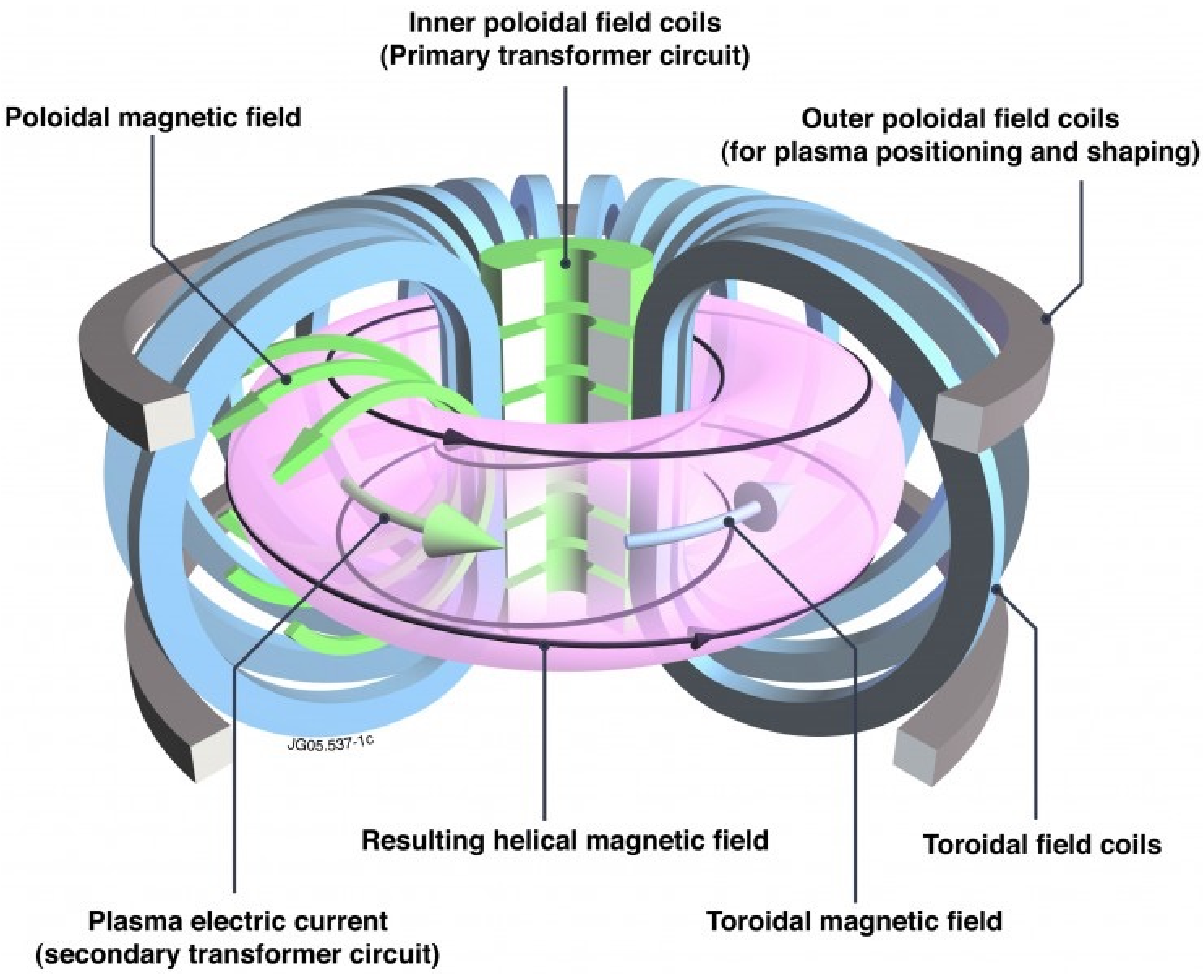

Fig. 2: Principle of the tokamak. The hot plasma is confined by a superposition of the toroidal and the poloidal magnetic fields. The first is generated by external magnetic field coils, the second by the electrical current induced in the plasma. Due to the transformer action to induce the current in the plasma, the tokamak is a pulsed device by definition (picture EUROfusion).

Hitherto, the highest fusion performance (16 MW) has been achieved in the Joint European Torus, JET, world's largest tokamak [3]. Also the international ITER experiment [4] – a collaboration of China, Europe, India, Japan, Russia, South-Korea and the United States – is based on the tokamak concept. ITER is expected to have first plasma around the middle of the next decade and is designed to achieve fusion power generation of about 500 MW, using 50 MW of external input power. ITER will not deliver any fusion electricity and will therefore be succeeded by DEMO, the first Demonstration Fusion Power Plant.

EUROfusion and the European Fusion Roadmap

Europe has drafted an elaborate plan to achieve the milestone of fusion electricity demonstration in DEMO by the middle of the century. In this so-called Fusion Roadmap [5], eight important missions have been defined, which can be grouped into:

1. Risk mitigation for ITER
2. (Pre-) Conceptual Design of DEMO
3. The stellarator as back-up strategy

Fusion research in Europe is coordinated by EUROfusion, a consortium of 29 National Fusion Laboratories from 27 countries, plus Switzerland, along with over 100 Universities, groups and industries that are acting as Linked Third Parties to the National labs.

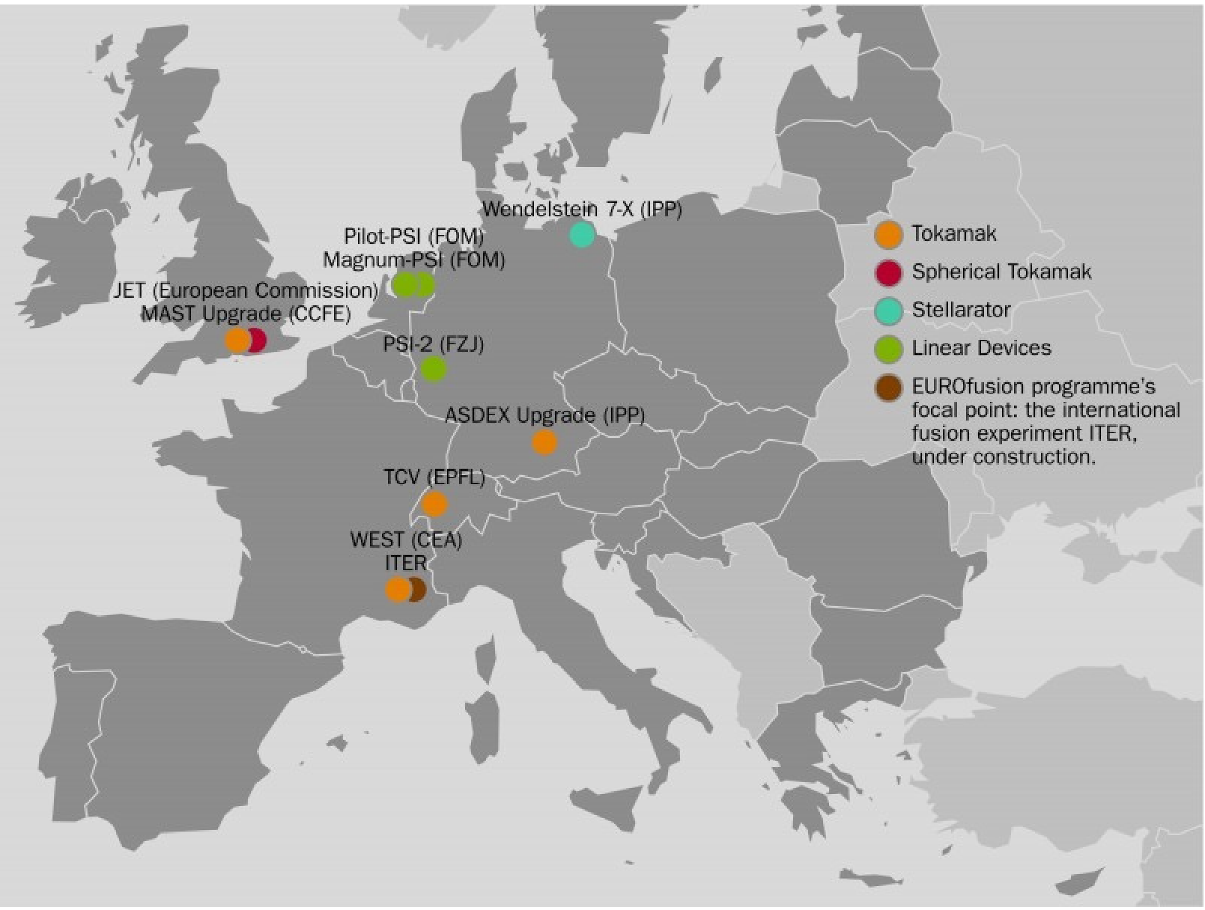

Fig. 3: Dark coloured countries are involved in the EUROfusion consortium (Sweden is only partly shown, and Finland is also involved). A large fraction of the EUROfusion research is concentrated on the various devices indicated with coloured dots. (Picture EUROfusion).

The fusion community is confident that ITER will work and reach its full performance and all of its objectives. However, there are open research issues that, if better understood, can help ITER to optimise its research plan. It is no surprise that there are even more open issues with respect to the design of the DEMO reactor. These are largely related to the very hostile environment with strong plasma-wall interaction and high fluxes and fluences of neutrons and gammas emerging from the hot plasma. In the remainder of this paper the reader will be guided through a few of the main physics challenges in the fusion roadmap.

The choice has been made to focus on items 1 and 3 above. Item 2 is linked to the DEMO design and preparation, and is more technology-oriented than the other two items, although it comprises challenging and interesting issues such as developing neutron-resistant materials, achieving tritium self-sufficiency, intrinsic safety, integrated DEMO design and competitive cost of electricity [6], which could be addressed in a future paper for EPN.

Risk mitigation for ITER (and DEMO)

The temperature of the fusion plasma in ITER (and also in DEMO) must be about 10-20 times higher than that in the core of the Sun, for colliding particles to have sufficient energy to fuse. Because there are strong temperature-, density- and current density gradients, the plasma is prone to develop microscopic instabilities (turbulence) as well as macroscopic magnetohydrodynamic instabilities, which degrade the plasma performance. The macroscopic instabilities can potentially completely destabilise the plasma, leading to a disruption, which ends the plasma state and leads to strong forces into the surrounding vacuum vessel due to induced halo currents [2]. So plasma scenarios need to be developed in which the performance is ramped up in a controlled way and in which instabilities are actively controlled. An excellent external 'knob' to control magneto-hydrodynamic instabilities is the injection of radio waves at the place of the instability. The radio frequency injected is either resonant with the local electron or ion cyclotron frequency or one of its higher harmonics. The cyclotron frequency $\omega_{ce} = eB/m_e$ for electrons and $\omega_{ci} = ZeB/m_i$ for ions, with e the electron charge, $m_{e,i}$ the electron or ion mass, Z the charge number of the ion, and B the magnetic field strength, which by itself is monotonically decreasing from the inside to the outside of the plasma column. This stabilisation method can act either on the electrons or on the ions in the plasma. By tuning the radio frequency waves onto the cyclotron resonance, it is possible to drive a local current near the position of the island which, dependent on its direction, can either reduce or amplify the instability. Another possibility to act on the plasma is the injection of powerful beams of neutral particles (typical energies in ITER ~1 MeV).

ITER will bring fusion physics into a new regime: The alpha particles carry 20% of the generated fusion power, which implies that at the highest ITER performance (fusion power/input power = 10), the self-heating by thealpha particles is twice the the external input power. This has a large effect on the way the plasma can be controlled. Only localized heating methods, with a high power density, like cyclotron heating can outweigh the alpha particle heating, and can therefore be used for efficient plasma control. Additionally, new effects can occur as the energetic alpha particles can interact with instabilities, which might lead to too high losses of fast particles. Many of these effects can be studied in already in present devices by mimicking alpha particles by fast ions that are externally injected,, but the ultimate proof-of-the-pudding of alpha-particle physics will be in ITER.

Achieving a high performance plasma is not the only challenge: By far the largest quest for the fusion researchers is to solve the heat exhaust problem. Namely, the power generated in the core of the plasma needs to be exhausted in a small part of the reaction chamber called the divertor. In ITER, the neutrons, more or less uniformly, deposit a total of 400 MW in the blanket structure surrounding the vacuum chamber. But about 90% of the remaining exhaust power of about 100 MW is convected towards the divertor, leading to a steady state heat load on the divertor components in ITER of 10-20 MW/m^2. These are power densities that are close to those at the surface of the Sun! The challenge of finding a proper solution beyond ITER is largely going into two directions: development of (new) plasma-facing materials that are more robust against the plasma-wall interactions as well as developing new magnetic geometries for the divertor in which the peak heat load is distributed over a larger surface. With respect to the latter direction: options that are being studied in Europe are the snowflake divertor in the Swiss TCV tokamak [7], the Super-X divertor in the British MAST-Upgrade tokamak [8] and liquid materials divertors in a number of specific experiments.

Plasma regimes of operation (mission1) and *Heat-exhaust systems* (mission 2) [9] in the fusion roadmap are tightly interlinked. This is illustrated by the following. Originally most tokamaks in the world utilised carbon tiles as main plasma-facing components and carbon-fibre composites (CFC) in the divertor, as this material is very strong and can withstand high temperatures up to about 1200°C. Carbon is also a relatively light atom and does not pollute the plasma too much when it enters (since the plasma is quasi-neutral, each impurity ion with charge number Z pushes out Z hydrogenic ions, leading to fuel dilution). However, carbon has two important drawbacks: 1) it forms dust, and 2) it binds with hydrogen. The effect of both is that in a machine operating with ^{3}T (like ITER) after a short time the whole tritium inventory is immobile due to retention in the carbon dust and carbon plasma-facing components. This implies opening and cleaning the machine and subsequently separating the tritium from the dust. It is for this reason that about 10-15 years ago a deliberate choice was made in Europe to switch to full metal machines. The German ASDEX-Upgrade [10], has gradually changed the wall material from full carbon to full tungsten. JET (see Fig. 4), has been modified in a single shutdown from a carbon machine to a device with beryllium walls and a tungsten divertor (exactly the same materials as will be employed in ITER) [11]. The WEST superconducting tokamak in France is presently being changed into a full tungsten device able to run long plasma pulses. Tungsten has a high melting point of 3422°C, but recrystallisation becomes important above 1200°C. The result of a few years of operation of ASDEX-Upgrade with a full tungsten wall and JET with the ITER-like wall is that the hydrogen retention has been reduced by a factor of ~15, which is sufficiently good for ITER [12]. However, it turned out to be much more challenging to achieve a high plasma performance due to influx and accumulation of tungsten in the plasma core, which – as sketched above – leads to considerable fuel dilution. This can be avoided by using special tricks as central plasma heating (with radio frequency waves), surrounding the plasma by a seeding gas and controlling instabilities at the plasma edge to purge the tungsten out of the plasma [13]. This shows the rather intricate interplay between reaching a high plasma performance and finding proper solutions for the plasma heat exhaust geometry and material choices.

Apart from the integrated research of plasma-wall interaction in tokamaks, new materials are constantly being developed and tested in linear plasma devices, like MAGNUM-PSI, Pilot-PSI and JULE-PSI, in which the materials can be exposed to plasma fluxes and fluences that are reminiscent to those in ITER [14]. In the linear machines, the region where the plasma interacts with the material is much better accessible for diagnostics compared to a tokamak, which makes these experiments very flexible and efficient.

The stellarator as back up strategy

Undoubtedly, the tokamak has the simplest design of the relevant confinement devices. Because it also has the best performance, international research has largely concentrated on this line since the 1970's. Besides its scientific successes, the tokamak has a number of drawbacks. Firstly, it is a pulsed device due to the fact that the plasma current is induced by a transformer. Secondly, the tokamak is prone to current-driven instabilities and disruptions that necessitate active control tools for a stable operation, as outlined above.

There is a second magnetic confinement device in which the confining magnetic field is completely generated by external field coils: the stellarator. The stellarator is in principle current free and, hence, the device is intrinsically more stable. But every advantage comes with a disadvantage: the design and construction of the stellarator is much more complex than that of tokamak (see Fig. 5), and this is the main reason why it is generally lagging behind the tokamak [15].

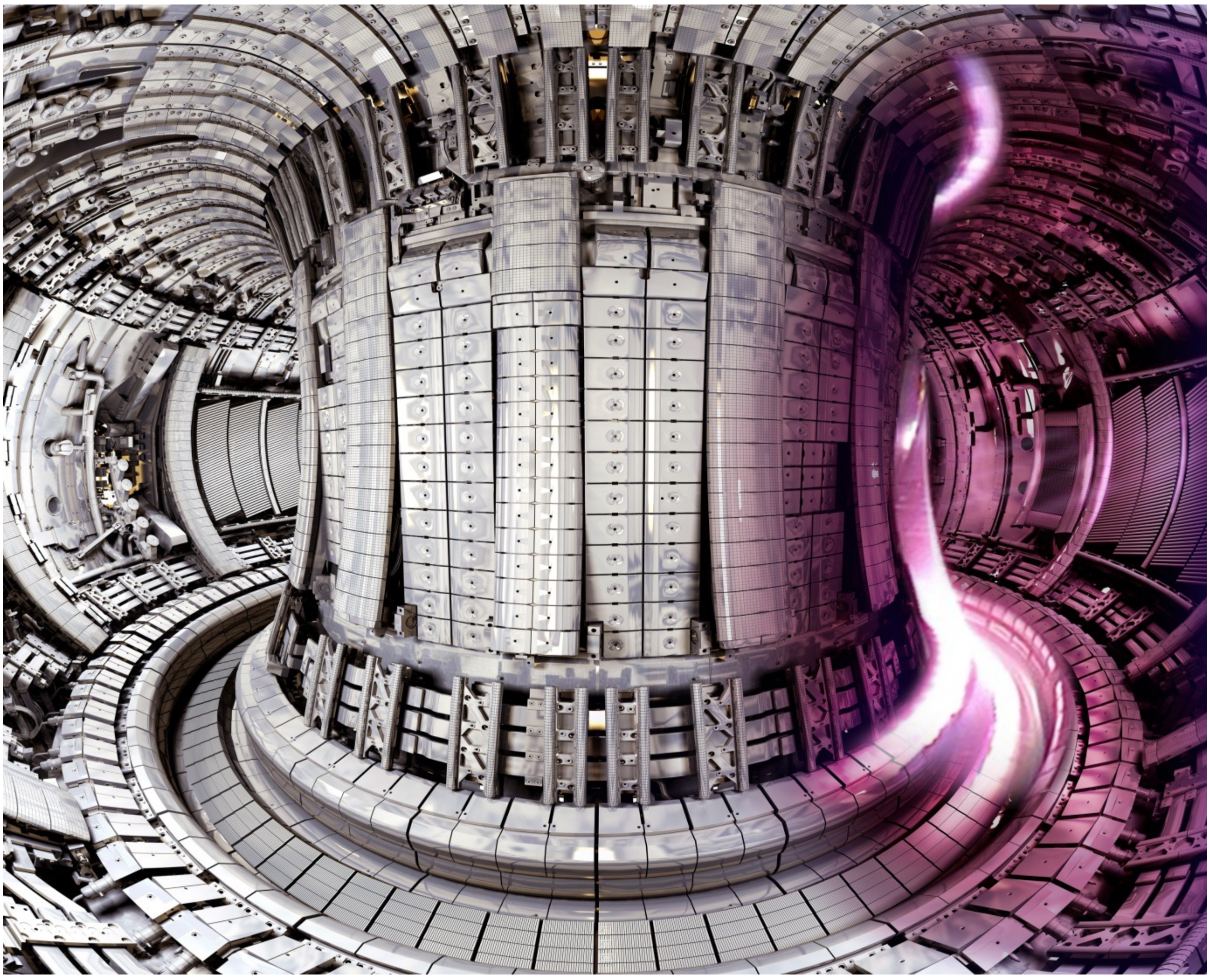

Fig. 4: Composite photograph of the JET vacuum chamber. By far most of the plasma wall is made of beryllium-coated Inconel tiles. The bottom of the machine, the divertor, is partly made of bulk tungsten and partly of tungsten-coated carbon-fibre composites. On the right hand side a picture of a hot plasma is overlaid. The brightest radiation (Balmer$_\alpha$) comes from the plasma edge (where most ionisation processes occur due to interaction of the hot plasma with the cold neutral gas surrounding it) and from the plasma-material interaction in the divertor. There is almost no Balmer$_\alpha$ emission from the fully ionized hot plasma core.

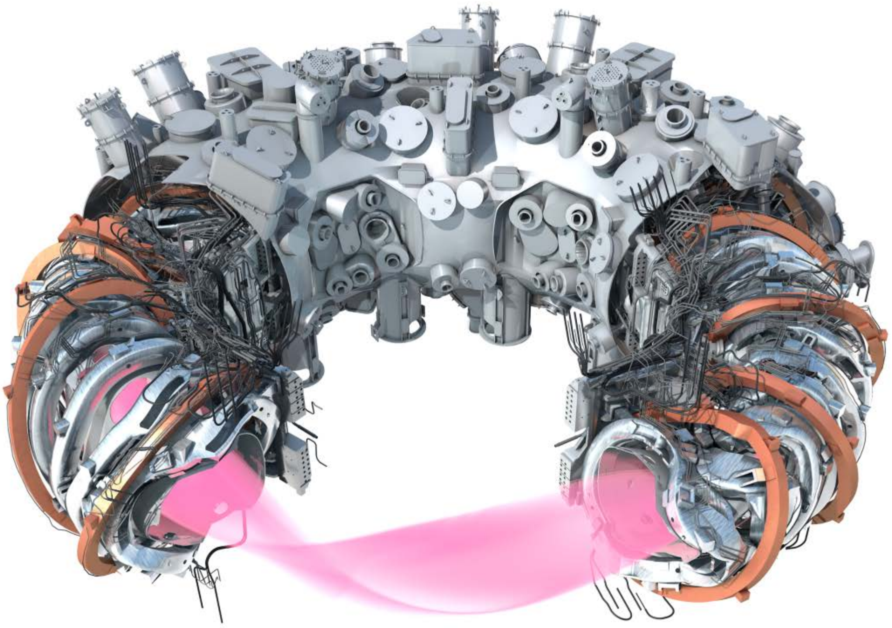

Fig. 5: Schematic drawings of the Wendelstein 7-X stellarator, showing the complexity of the device with the magnetic field coils, cooling channels, vacuum vessel and cryostat (copyright Max- Planck-Institut für Plasmaphysik)

Nevertheless, stellarator research has entered a new era: On 10 December 2015, the super-conducting Wendelstein 7-X device with its optimised magnetic configuration, located in Greifswald, Germany, and with a diameter of 16 m (see Fig. 5) has been taken into operation. Angela Merkel initiated on 3 February 2016 the first hydrogen plasma, which had already an electron temperature of 8 keV. Research in Wendelstein 7-X will show the viability of this concept and its potential for a future fusion power plant.

Concluding remarks

In this brief paper it has only been possible to describe a small fraction of the European research in nuclear fusion, and in doing that even only the tip of the iceberg could be discussed. There are still many scientific and technological challenges in fusion research, ranging from a very fundamental nature to more applied issues. Apart from that it is a very interesting and rewarding discipline to work in, it has the additional prospect that it is contributing towards a solution to the world energy and climate problem.

About the Author

Tony Donné is Programme Manager of the EUROfusion consortium, a position he has held since June 2014. He obtained his PhD degree (1985) at the Free University of Amsterdam. Most of his scientific career was devoted to research in the field of high-temperature plasma diagnostics. From 2009 – 2014 he was heading the fusion research department of the Dutch Institute for Fundamental Energy Research.

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