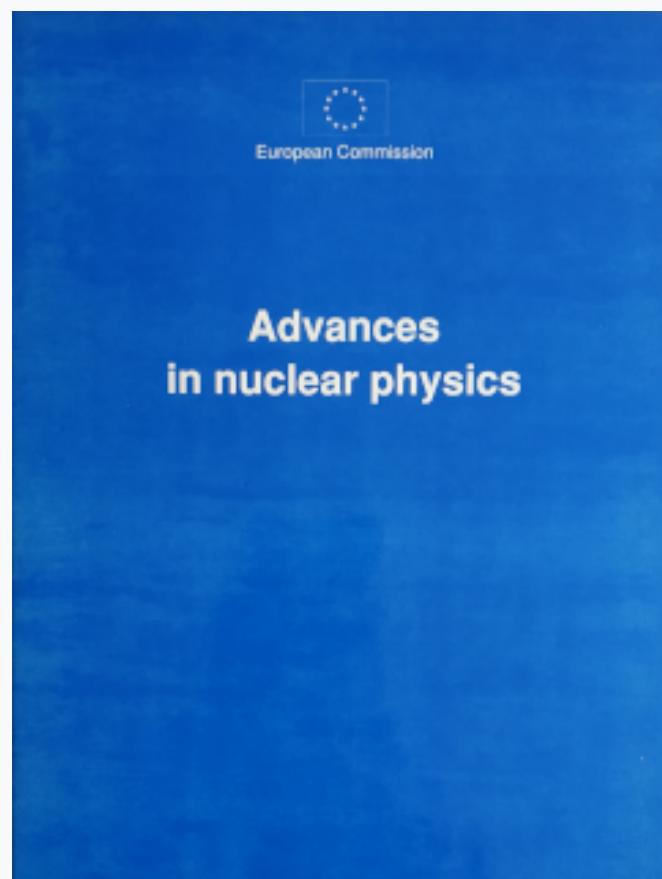


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Physics contributions from IPP Garching to the fusion program

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Introduction

Nuclear fusion research is focused on the development of physics and technology for controlled utilization of the fusion reaction $D + T \rightarrow {}^4\text{He} + n$ which releases an amount of 17.6 MeV of energy. In the thermonuclear approach to fusion a DT-plasma is heated to high temperatures such that a sufficient reaction rate is provided by the high energy tail of the Maxwellian distribution of the plasma particles. Ignition of the plasma is reached if the energy gain by α -particles, which is 3.5 MeV per reaction, is in excess of the energy loss of the plasma due to imperfect energy confinement. For ignition the plasma ion temperature T_i has to be larger than 10 keV and the product $n\tau_E$ of plasma density n and energy confinement time τ_E must simultaneously exceed $4 \times 10^{20} \text{ m}^{-3}\text{s}$, i.e. $n\tau_E T_i > 4 \times 10^{21} \text{ m}^{-3}\text{skeV}$.

Confinement of the hot plasma is achieved by magnetic fields, utilizing the Larmor motion of charged particles perpendicular to the field direction. Toroidal field structures are favoured because no plasma losses occur along the field lines. However, from equilibrium considerations it can be seen that a purely toroidal field is not sufficient to confine a plasma. There must be an additional poloidal magnetic field for stable confinement. There are two favourite types of toroidal devices, tokamak and stellarator, which basically differ in their way of how to produce the poloidal field component. In a tokamak the toroidal magnetic field is produced by external coils, the poloidal field arises from a current flowing in the plasma in the toroidal direction, the current being inductively driven by a transformer. In a stellarator the confining magnetic field is completely produced by external coils.

Much progress has been achieved since the beginning of fusion research, e.g. visualized by the improvement of the fusion product $n\tau_E T_i$ from $< 10^{18} \text{ m}^{-3}\text{skeV}$ in 1970 to $0.9 \times 10^{21} \text{ m}^{-3}\text{skeV}$ in the Joint European Torus (JET) and $1.1 \times 10^{21} \text{ m}^{-3}\text{skeV}$ in the Japanese Tokamak 60 Upgrade (JT-60U) in 1993. Ignition is expected for the next step device, the International Thermonuclear Experimental Reactor (ITER). However, some physics problems still remain to be solved. The basic mechanisms of particle and energy transport perpendicular to the magnetic field are not yet understood and the empirical transport coefficients are - at least for the electron component of the plasma - in excess of predicted neoclassical values. Transport is therefore termed anomalous. Extrapolation from smaller to larger devices and to a reactor is therefore based on empirical scaling laws which are obtained from data bases carefully supplied by experiments of various size.

Plasma heating systems are well developed (ohmic heating by an inductively driven plasma current, auxiliary heating by neutral beam injection (NBI), electron and ion cyclotron resonance heating (ECRH and ICRH)) but large efforts are made to drive plasma currents non-inductively by proper application of auxiliary heating, either by input of toroidal momentum or by preferential heating of particles moving in a specific toroidal direction. The internal plasma current driven by the gradient of the plasma pressure (bootstrap current) may be influenced by tailoring the radial pressure profile via local power deposition. The need for non-inductive current drive arises from the wish for continuous operation, which is inherent to a stellarator but is limited for a tokamak by the finite flux swing of the current driving transformer. Stellarators are excellently suited for physics studies of non-inductive current drive and bootstrap current which are not masked by large inductive currents.

Impurities are produced at surfaces in contact with the plasma. If they enter the core plasma they dilute the fusion fuel and lower the plasma temperature by radiating energy $\sim Z^2$ away, both being detrimental for the fusion product. Additionally, in a reactor the helium ash and the fusion energy must be removed in a controlled way. Local power flux densities at plasma facing components must thereby be kept at tolerably low level ($< 10 \text{ MW/m}^2$). A promising concept for impurity control as well as particle and energy exhaust is the so called divertor, which is a key element of the IPP tokamak line.

At the Max-Planck-Institut für Plasmaphysik (IPP) in Garching both types of toroidal confinement devices, tokamak and stellarator, are investigated. The tokamak program is concentrated on the realization of a reactor relevant divertor and the study of reactor like plasma boundary physics. The advanced stellarator program is based on physics optimization of stellarator configurations and its realization by modular field coils. Apart from device specific investigations both types of experiments contribute to basic questions of fusion research, such as transport, confinement and current drive in toroidal devices.

The IPP divertor tokamak program

The IPP tokamak program concentrates on the development of the divertor concept. In a divertor configuration field lines at the periphery of the plasma confinement region are "diverted" by additional magnetic coils

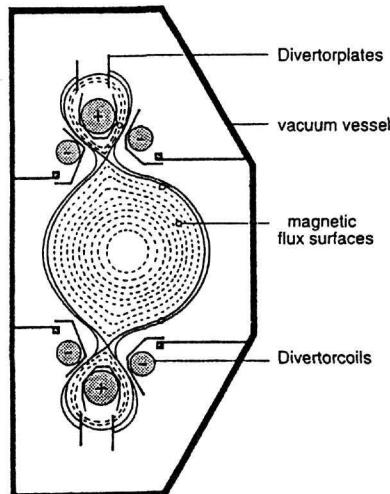


Fig. 1: Poloidal cross section of the ASDEX closed divertor configuration

and led to target plates far away from the main plasma (fig. 1). The divertor prevents direct contact of the hot bulk plasma with enclosing surfaces. The temperature drops from about 100eV at the plasma edge to about 10eV at the divertor target plates. Consequently surface erosion and impurity production rates are reduced. Impurities released at the target plates are ionized with high probability in the divertor region and are swept back to the plates before they can contaminate the main plasma. The build up of neutral pressure in front of the plates eases particle removal (impurities, excess fuel and helium ash) by pumping.

Realization of the divertor principle was successfully demonstrated in the Axially Symmetric Divertor EXperiment (ASDEX, device parameters: large radius $R = 1.65\text{m}$, plasma radius $a = 0.4\text{m}$, magnetic field $B_0 \leq 2.8\text{T}$, plasma current $I_p \leq 0.5\text{MA}$).

Extremely clean plasmas could be produced.

In addition, a regime of improved plasma confinement - the so called H-mode - was surprisingly discovered¹. In this high confinement regime a transport barrier develops at the plasma edge improving the energy confinement time by typically a factor of 2 for divertor tokamaks. From todays experience, a divertor configuration eases the occurrence of the H-mode and yields a higher confinement improvement than a limiter configuration but it is not necessarily required. The underlying physics of the H-transition is not yet understood. Empirically it is related to the development of sheared (i.e. radially varying) poloidal plasma rotation. This sheared flow may radially decorrelate plasma fluctuations, which are assumed to be responsible for anomalous transport. Theories relying on the plasma current as a basic ingredient for the H-mode can no longer be supported since the first discovery of an H-mode in a stellarator (see below).

The issue of ASDEX-Upgrade² (AUG, $R = 1.65\text{m}$, $a = 0.5\text{m}$, elongation $\kappa = 1.6$, $B_0 \leq 3.9\text{T}$, $I_p \leq 2\text{MA}$), which became operational in 1991, is to test the divertor concept under reactor like conditions but on a smaller scale. AUG is adjusted to reactor requirements in respect to coil geometry, plasma configuration, divertor shaping and plasma boundary parameters. In ASDEX, which was shut down in 1990, the divertor coils were located close to the plasma in

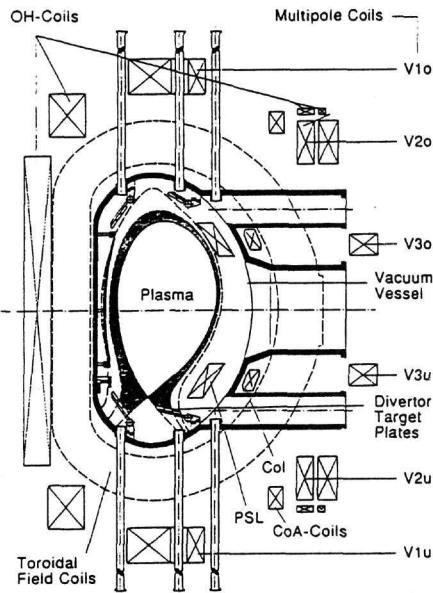


Fig.2: Poloidal cross section of ASDEX Upgrade with an open divertor configuration

separate "closed" divertor chambers inside the vacuum vessel. In a burning reactor the divertor configuration has to be "open" with the divertor coils installed outside the vessel (fig. 2). To simulate reactor like plasma edge conditions, in particular a high energy flow across the plasma boundary into the divertor, AUG is equipped with sufficiently high heating power. AUG investigates energy transport to the target plates and the wall, impurity control, hydrogen recycling and helium pumping and will test target plate and wall materials. In view of extremely high power fluxes into the divertor expected for ITER, radiative edge cooling by controlled impurity injection appears to be necessary and will be studied in AUG for control of the edge temperature and the divertor power load.

The IPP advanced stellarator program

Stellarators offer a large variety of magnetic configurations by shaping the confining fields with external coils. A classical stellarator consists of a set of toroidal field coils and a set of helical windings (fig. 3). One of the milestones of stellarator research was the first net current free operation of the classical stellarator Wendelstein 7-A (W7-A, $R = 2\text{m}$, $a = 0.1\text{m}$, $B_0 \leq 3.5\text{T}$) at the IPP, which became possible by the availability of NBI and ECRH heating systems to replace ohmic heating³. In this operating mode only a small inductive current is applied to compensate for the pressure driven bootstrap current. Low current operation is a fundamental prerequisite for the full utilization of the flexibility inherent to the stellarator concept, because plasma currents perturb the external field structure and consequently its confinement properties. Furthermore, current free stellarators can operate continuously, they are not susceptible to current driven instabilities and to the loss of confinement due to sudden current quenches (disruptions). W7-A was shut down in 1985.

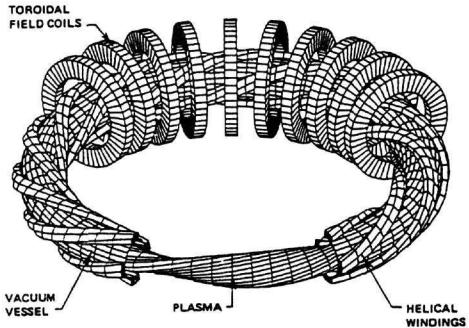


Fig.3: Toroidal and helical coil system of the classical stellarator Wendelstein 7-A

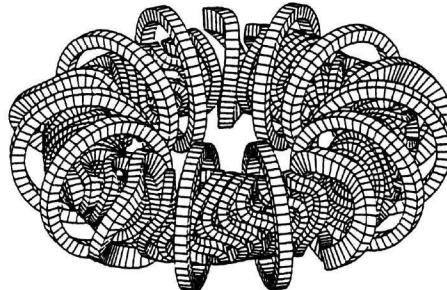


Fig.4: Modular coil system of the advanced stellarator Wendelstein 7-AS

Technical complexity because of the large helical coil system and unfavourable neoclassical confinement properties of classical stellarators led to the need for stellarator optimization⁴, which resulted in the IPP advanced stellarator concept with modular coil system. The modular design with a set of twisted but poloidally closed field coils, which provide both toroidal and poloidal field components, is best suited for realization of optimized stellarators and appears to be necessary in view of large size stellarator reactors. The principles of stellarator optimization are^{4,5}

- good magnetic surfaces of the vacuum field
- low Shafranov shift and good finite beta equilibrium
- good magneto-hydrodynamic (MHD) stability properties
- small neoclassical transport losses
- small bootstrap current
- good confinement of α -particles
- good technical feasibility of a modular coil system

First representative of the advanced stellarator line is the modular Wendelstein 7-Advanced Stellarator⁶ (W7-AS, $R = 2\text{m}$, $a = 0.18\text{m}$, $B_0 \leq 3\text{T}$), which is in operation since 1988 (fig. 4). W7-AS has successfully demonstrated the technical feasibility of a modular coil system. Figure 5 shows the excellent agreement between calculated and measured vacuum flux surfaces. W7-AS is partly optimized: The Shafranov shift - which is a pressure induced radial outward shift of the plasma column limiting the maximum achievable pressure - is reduced by a factor of 2 as compared to the classical stellarator W7-A. Experimentally the reduction could be quantitatively confirmed by the radial displacement of soft X-ray emission from the plasma center⁴. Objectives of W7-AS are confinement and transport in a low shear stellarator, heating and current drive by ECRH and NBI, neoclassical effects such as bootstrap current, and plasma edge physics with respect to the divertor potential of stellarator

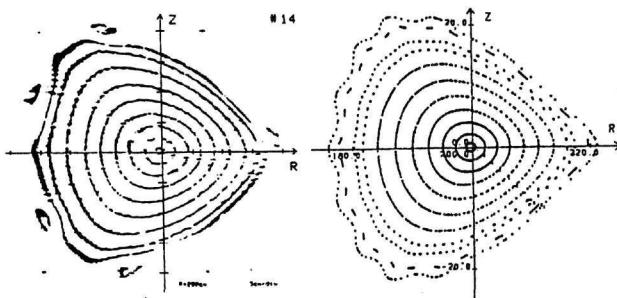


Fig.5: Measured (left) and calculated (right) flux surfaces in W7-AS

specific boundary topologies. An unexpected result of W7-AS was the first discovery of the H-mode in a stellarator⁷. Although the improvement factor in global confinement is only about 1.2, the W7-AS H-mode shows all basic features known from tokamaks.

Full stellarator optimization will be realized in the next step stellarator Wendelstein 7-X⁸ (W7-X, $R = 5.5\text{m}$, $a = 0.53\text{m}$, $B_0 \leq 3\text{T}$). For example, by optimization the bootstrap current in W7-X will only be 10% as compared to an equivalent tokamak, and the Shafranov shift will be reduced by a factor of 8 as compared to W7-A, resulting in largely improved equilibrium and stability properties (maximum relative plasma pressure $\beta = p/(B^2/2\mu_0) = 5\%$ as compared to 2% in W7-AS). The coil system will be constructed with superconducting technology allowing for steady state operation. The conceptual design of W7-X is completed and has been approved by EURATOM. Construction will probably start in the near future.

Summary

Both types of toroidal fusion devices are investigated at the IPP Garching, tokamak and stellarator. The tokamak program, currently represented by the ASDEX-Upgrade tokamak, is concentrated on the realization of an open divertor configuration and its test under reactor like plasma boundary conditions. The open divertor matches the requirements of particle and energy exhaust as well as impurity control in a reactor. The advanced stellarator program is based on physics optimization of stellarators. The proof of principle has been successfully realized with the partly optimized advanced stellarator W7-AS. A fully optimized stellarator, W7-X, is being prepared and will test the improved stellarator concept under reactor relevant plasma conditions.

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