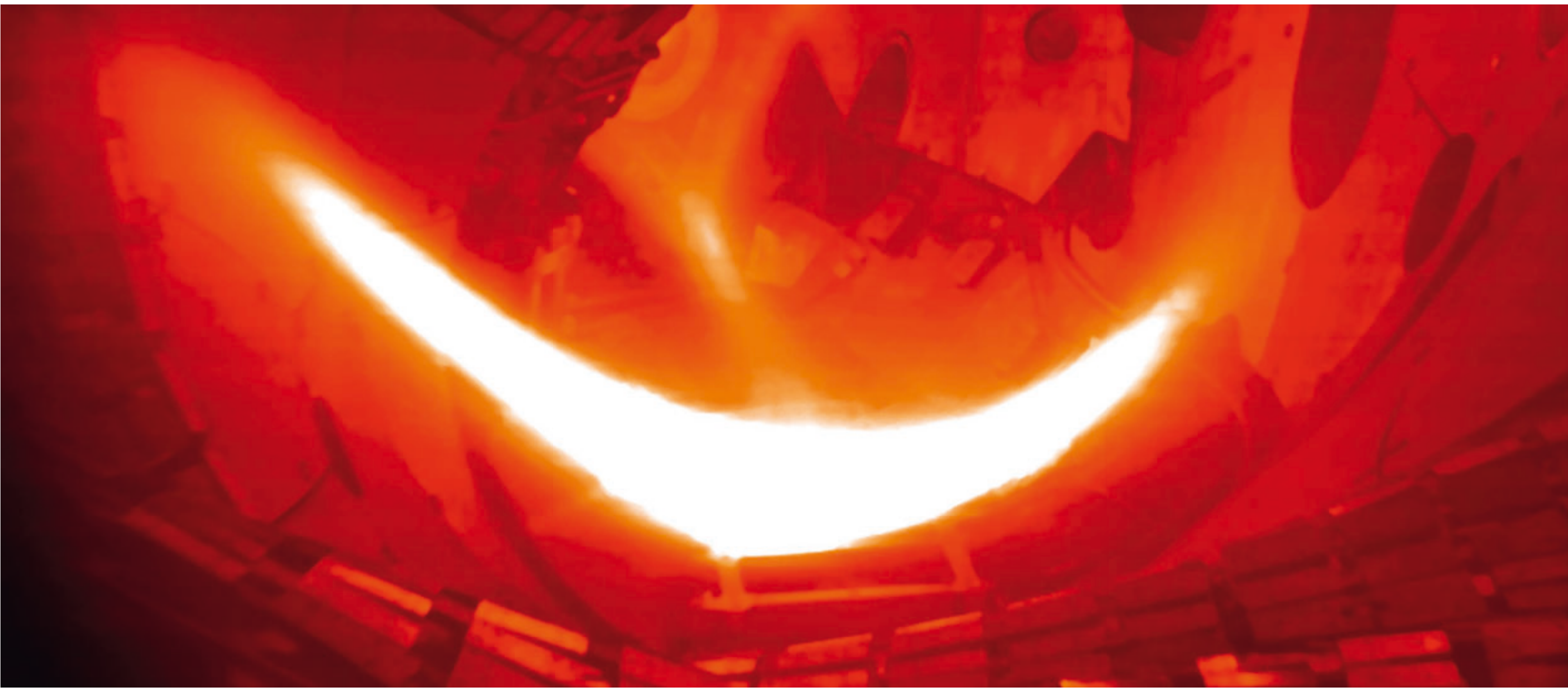
Max-Planck-Institut
für Plasmaphysik

Scientific Report 2015 - 2016



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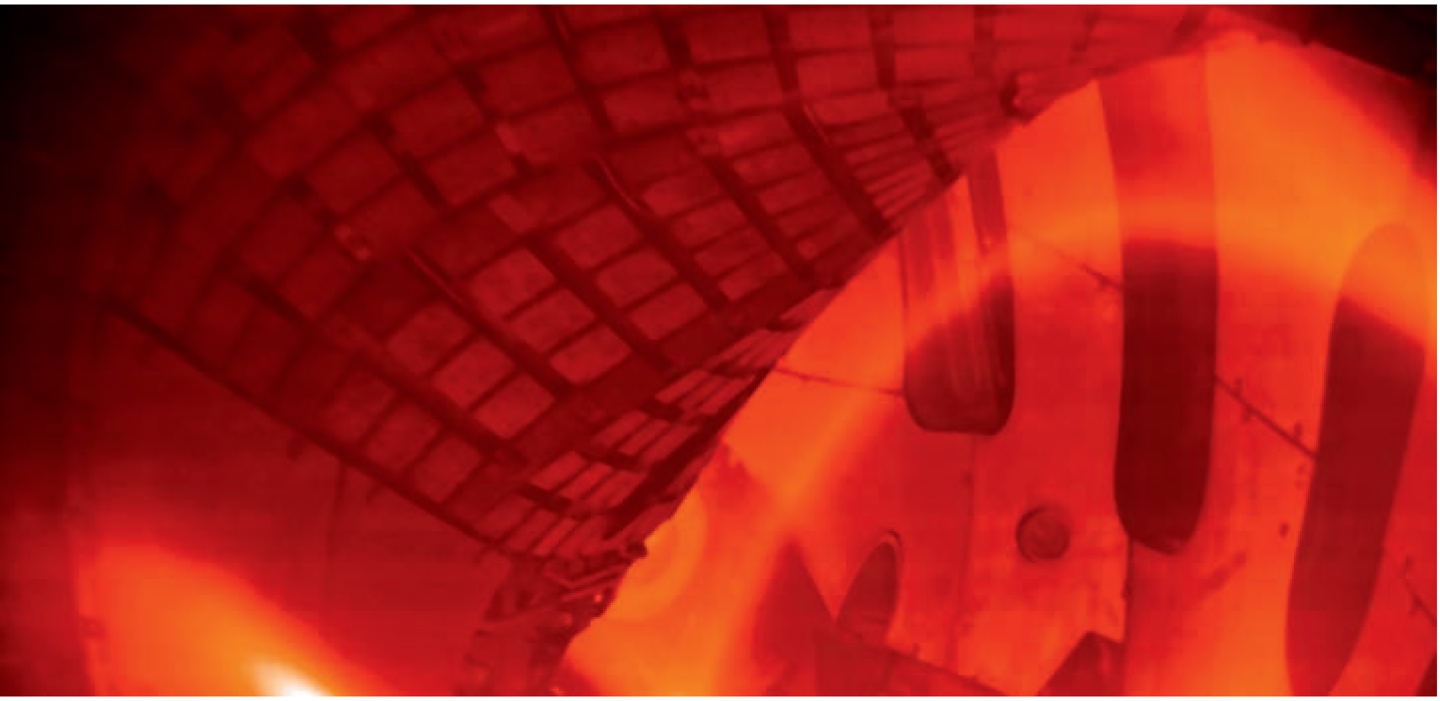

On 3 February 2016 the Wendelstein 7-X stellarator produced its first hydrogen plasma.

Max-Planck-Institut
für Plasmaphysik

Scientific Report

January 2015 - September 2016

The Max-Planck-Institut für Plasmaphysik is an institute of the Max Planck Gesellschaft, part of the European Fusion Programme and an associate member of the Helmholtz-Gemeinschaft Deutscher Forschungszentren.

Photo: IPP, Silke Winkler

Wendelstein 7-X activities in 2015 focussed on the commissioning of the device and the preparation of the first plasma operation. Major milestones were the successful cool-down of the coils and their support structure inside the cryostat to 4 kelvin, the first operation of the magnets up to a field of 2.5 tesla, and the proof that the magnetic field structure is very accurate with deviations from design configuration less than 10^{-5}. The first helium plasma operation was achieved on December 10th, 2015. The event was followed by numerous guests from the world-wide science community – most of them by video broadcast – and the national and international press. The first hydrogen plasma was produced on February 3rd, 2016, during an inauguration ceremony in the presence of German Federal Chancellor Dr. Angela Merkel. The first experimental campaign, lasting until March 10th, 2016, was very successful. Plasma parameters were already very impressive – 100 million degrees for the electrons and 20 million degrees for the ions – and discharge lengths of a few seconds were also achieved. The success of the first experimental campaign greatly exceeded our initial expectations. The original aim was to perform an integral commissioning of the Wendelstein 7-X plasma operation. However, swift progress allowed many in-depth physics studies. Meanwhile, with the installation of the first divertor Wendelstein 7-X enters the next completion step.

In the report period from January 2015 to the end of September 2016, the ASDEX Upgrade tokamak was operated for two campaigns, again with roughly half of the operational time being devoted to the Medium Sized Tokamak campaign of the EUROfusion consortium. There were a large number of interesting results in support of ITER and DEMO, but also concerning fundamental high-temperature plasma physics.

As of the 2015 campaign, newly developed three-strap ICRH antennae were used to heat the plasma, showing the predicted reduced influx of tungsten from the antenna protection structure. This very positive result opens the way to using ICRH in a DEMO reactor, which will have to cover the first wall with high-Z materials. Another DEMO-relevant result was the achievement of fully non-inductive tokamak discharges for more than 30 energy confinement times by using a mix of ECCD, NBCD and the self-generated bootstrap current, the latter being responsible for about 50 per cent of the total current. The normalized parameters were close to those envisaged for the EU DEMO, indicating that with the completion of the ECCD system upgrade, this scenario can be further optimized to meet DEMO requirements.

Concerning the preparation of ITER operation, a comprehensive campaign using helium as working gas demonstrated the possibility of operating ITER in ELMy H-mode already in the non-nuclear phase. It was also shown that in deuterium, by lowering the collisionality and adjusting the plasma shape, ELMs can be fully suppressed by using the RMP coils. These experiments help to understand the underlying physics mechanisms, in particular the response of the plasma to the externally applied field – important for predicting the effect of RMPs on ELMs in ITER.

A new highly performant CXRS diagnostic advanced our understanding of the mechanism triggering the transition from L- to H-mode confinement. In contrast to findings from other tokamaks, on ASDEX Upgrade the transition is mainly induced by neoclassical ExB shear flows; zonal flows are of minor importance. It was also possible to explain the dependence of the power threshold of the L-H transition on isotopic mass, magnetic field strength and density. An improved model for the pedestal stability allowed the influence of various parameters, such as impurity seeding and the wall material, on confinement to be described. These two important findings substantially improve the capabilities to predict confinement in ITER and DEMO. For better prediction of divertor power loads during ELMs in ITER, infra-red measurements from different devices were combined to a scaling expression. This scaling leads to more optimistic predictions for ITER, especially with mitigated ELMs at high plasma density.

The new divertor manipulator system on ASDEX Upgrade was used to study the power-handling capabilities of tungsten divertor target plates under repetitive transient ELM power loads in order to cross-check earlier unexpected results from JET. The findings allowed a refined analysis of the JET results removing the apparent inconsistency in the earlier interpretation. While the proof-of-principle for the applicability of tungsten-fibre-reinforced tungsten as plasma-facing material was shown in preceding years, new production routes now allow fabrication of larger samples for mechanical testing.

IPP again significantly contributed to the JET programme: Physics studies on power and particle exhaust by nitrogen seeding were continued. In this respect, IPP also investigated the ammonia production and retention in nitrogen-seeded discharges. Very exciting results have been found regarding the effect of helium on plasma performance. This work to a large extent benefited from corresponding studies in ASDEX Upgrade and diagnostic development. Furthermore, a long-term project focussing on major improvements of data interpretation based on Bayesian methods was finalized. JET operation was also supported by control room expertise for ICRH heating, thermography and bolometry interpretation, SXR analysis and diagnostic coordination. Modelling of SOL transport investigated the role of electric fields for L-H transitions.

The direct contributions to the ITER project continued: The ELISE test facility successfully demonstrated the first one-hour pulse in deuterium at low pressures and with an acceptable electron-to-ion ratio. Caesium management in the source could also be improved. For the ECRH upper launcher various scenarios for the deposition profile width were investigated, leading to new insights for beam-broadening effects. Within the Framework Partnership Agreement (FPA) for the ITER diagnostic pressure gauges, the system level design was completed with the definition of the baseline design. The FPA on the development of the ITER bolometer diagnostic completed the detailed project planning and started system level design activities. A first version of the plasma control system simulation platform for ITER was released for community use and the preliminary architecture of the plasma control system was defined. Furthermore, as part of contracts with ITER, theoretical investigations for runaway electrons and halo currents were performed.

In the theory divisions of IPP many interesting and important results have been achieved. For example, detailed theoretical and numerical studies have allowed an improved understanding of controversial issues in transport physics. It was thus possible, for instance, to show that the turbulent diffusion of heavy impurities such as tungsten can be maximized for a given heating power by tuning the heating schemes in such a way that the electron and ion heat fluxes are of comparable magnitude.
Stellarators could be shown to benefit from novel turbulence-reducing mechanisms that significantly improve the plasma confinement. On the more mathematical side, a variational framework for electromagnetic particle-in-cell codes was developed. Its exact discrete variational structure enables an exact preservation of certain constraints of Maxwell's equations. These are essential features for long-time kinetic simulations of tokamak or stellarator plasmas.

Scientific Director Sibylle Günter

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Tokamak Research

ASDEX Upgrade

Head: Prof. Dr. Arne Kallenbach

The ASDEX Upgrade experimental program is devoted to the preparation of ITER operation, research for basic physics understanding and the design of a prototype fusion reactor, DEMO. In 2015-16, ASDEX Upgrade operated nine months, in equal shares for IPP-internal and the new EUROfusion MST1 programs. The 2014 extensions, notably the new pair of 3-strap ICRF antennas, performed very well. End of May 2016, a vent started scheduled till the end of the year.

1 Overview

ASDEX Upgrade (AUG) operation was conducted from July 2015 till May 2016 with 90 experiment days and 1540 useful plasma discharges. 697 discharges were performed for the EUROfusion MST1 program, while 685 # covered internal proposals. The major enhancement for 2015, the new pair of 3-strap ICRF antennas performed very well, and the new B-coil AC power supply BUSSARD was finally extended with a set of 16 individual AC power supplies. The integration of both IPP internal and MST1 program proceeded well, while the parallel operation of AUG and TCV entered the planning boundary conditions with additional constraints.

1.1 Major Physics Results

Scientific highlights in plasma operation comprise the achievement of full ELM suppression by magnetic perturbations (MP), the development of a robust scenario for non-inductive operation up to a plasma current of 0.8 MA, and the as-predicted reduction of tungsten release by the new 3-strap ICRF antennas. The long sought-after explanation for the improved energy confinement with nitrogen seeding was found by careful data analysis of pedestal profiles in combination with stability calculations. Further light could be shed on the role of the neoclassical edge electric field on the L→H transition.

Full ELM suppression by MP at low pedestal collisionality was obtained in collaboration with DIII-D, where a triangularity dependence of the ELM suppression threshold had been detected in AUG similarity studies. Hence, full ELM suppression was obtained in AUG at elevated triangularity δ and a safety factor q_{95} around 3.7 with MP with a toroidal mode number n=2. ELM suppression is accompanied by a strong density pump-out, the energy confinement is moderately degraded resulting in H_{98}=0.9-0.95. Due to the required very low density, corresponding experiments can only be performed during the first 1-2 days after a boronisation.

By a combination of neutral beam current drive, NBCD, and localized electron cyclotron current drive, ECCD, at different radial positions, about full non-inductive operation could be achieved for seconds, where the self-generated bootstrap current provided about 50% of the total plasma current. These plasmas are situated close to the no-wall MHD stability limit in β_N. The safety factor q is kept above a minimum value of 1.5 in order avoid sawteeth and NTM with a helicity up to and including (m,n)=(3,2).

Major progress in physics understanding was achieved for the underlying reason of the energy confinement improvement during nitrogen or carbon (CD4) seeding with full-W first wall. The reduction of the high field side high density (HFSHD) pattern by the enhanced SOL radiation leads to an inward shift of the pedestal density profile due to changes in fuelling.

Since the temperature profile is anchored at the separatrix, this density profile shift leads to a modification of the pressure profile and hence also of the magnetic shear closely inside the separatrix. The better alignment of pressure gradient and shear results in a higher pedestal top pressure and global energy confinement. Further effects towards enhanced edge stability are a reduction of the bootstrap current by increased Z_{eff} and an amplification of the pedestal confinement improvement by the enhanced Shafranov shift, acting also positively via a shear modification. The different elements contributing to the edge pressure limits have been integrated in edge stability calculations, which nicely reproduce the experimental variations of the pedestal top pressure versus separatrix density and heating power.

Thanks to diagnostic improvements, investigations of fast E_r changes during and around the L→H transition shed further light on the underlying physics. Down to a time scale of 100 µs, the radial electric field is in line with the pressure gradient and described by the neoclassical prediction, meaning that the poloidal E×B drift just cancels the ion diamagnetic drift, leaving the ions more or less at rest. This suggests no large contribution from zonal flows to the radial electric field and to the L→H transition mechanism at least on the resolved timescale. Limit cycle oscillations exhibit an ELM-like behavior, where particles and energy are expelled during turbulent phases with weaker gradients and lower E_r. Toroidal field scans revealed that the important player for the L→H transition is in fact the sheared flow $v_{E\times B}$ and not E_r itself.

1.2 Machine Enhancements

The major enhancement for the 2015/6 campaign was the installation of the pair of 3-strap ICRF antennas. Main goal of this enhancement was a reduction of the tungsten release during ICRF operation due to a reduction of antenna frame mirror-current induced sheath electric fields. Indeed, a reduction of the ICRF-induced rise of the plasma W content by about a factor two in comparison to previous 2-strap antennas with W coated limiters could be demonstrated. The W influx from the antenna side limiters showed a corresponding reduction with some poloidal variation along the limiter.

The coverage of the heat shield with steel tiles had been cautiously proceeded with the installation of one additional, third row with alternating P92 and Eurofer steel tiles.

The reason for the caution was the servation of damaged graphite tile edges opposite to certain steel tiles after the 2014 campaign. The underlying mechanism could be identified as currents flowing in the support structure during disruptions. After stiffening the support structure and modifications of the electrical insulation, the steel coating proceeds with the installation of in total seven rows of steel tiles in 2016.
Preparation of the 2017 program was started with the call for participation in the IPP internal and MST1 programmes on July 29th. AUG restart is foreseen end of January 2017. Apart from a short 2-months vent planned for August/September, the remainder of 2017 will have plasma operation.

2 MST Activities

The AUG device is one of the pillars of the EUROfusion Medium-Size Tokamaks (MST) Task Force, together with the TCV experiment at the EPFL Swiss Plasma Center in Lausanne and the MAST-U experiment at CCFE in Culham.
AUG and TCV have been contributing to the 2015/16 MST campaign, while MAST-U is still under construction after a major upgrade. The top three main objectives of that campaign have been: (a) understand the effect of density versus collisionality and operation with an all metal wall for ELM mitigation/suppression with pellets and resonant magnetic perturbations; (b) Increase efficiency and understanding of methods for disruption mitigation or avoidance and runaway electrons control; (c) Increase the operational margin for ITER and DEMO relevant scenarios with high P_{sep}/R and tolerable target heat loads. The MST experiments in AUG have provided valuable contribution to all these major objectives and strong synergies between AUG and TCV have been developed, with shared design and execution of experiment. As an example, 20 experiments out of 47 MST1-AUG experiments had their counterpart on TCV.
In the 2015/16 campaign the MST part of the program covered 50% of the total experimental time of AUG (697 pulses). Altogether 268 scientists from EUROfusion members and international collaborators contributed to the MST1-AUG program with a significant fraction (up to 20%) coming from IPP.

ELM Control

The repetitive transient heat loads due to ELMs are a key challenge for fusion power plants and ELM control or small/no ELM is mandatory. A multi-machine database of peak type-I ELM energy fluence, $\varepsilon_{\parallel}$, developed on AUG and JET shows that within a factor of 3 ELM heat loads can be understood by a simple flux tube model and are proportional to the pedestal pressure, p_{ped} , times the machine minor radius. Interestingly, data of type-I ELM control at low collisionality, $\nu^*_{ped} \lesssim 0.4$, with resonant magnetic perturbations falls into the same trend as the natural ELM data and there is so far little indication that $\varepsilon_{\parallel}$ can be reduced by more than a factor of 3 at a given p_{ped}, suggesting that most of the reduction of the relative ELM energy loss, $\Delta W_{ELM}/\Delta W$, is due to the density pump-out. Indeed, a database study for n=2 MP ELM control in D shows that the $\Delta W_{ELM}/\Delta W$ correlates well with n_e^{ped} and less well with ν^*_{ped} . Whilst the density pedestal after a 30% pump-out can be restored using shallow ITER like pellet fuelling by doubling the fuelling rate the p_{ped} remains below the non-MP phase. MP ELM control at low ν^*_{ped} is very sensitive to the alignment of the perturbation to the magnetic field. Only a dominant edge kink plasma response localized close to the X-point leads to a reduction in ELM energy loss. These plasma response calculations have been validated against measurements at the mid-plane and top of the plasmas during toroidal rotation of the perturbation field at fixed phase between the upper and lower coil set.

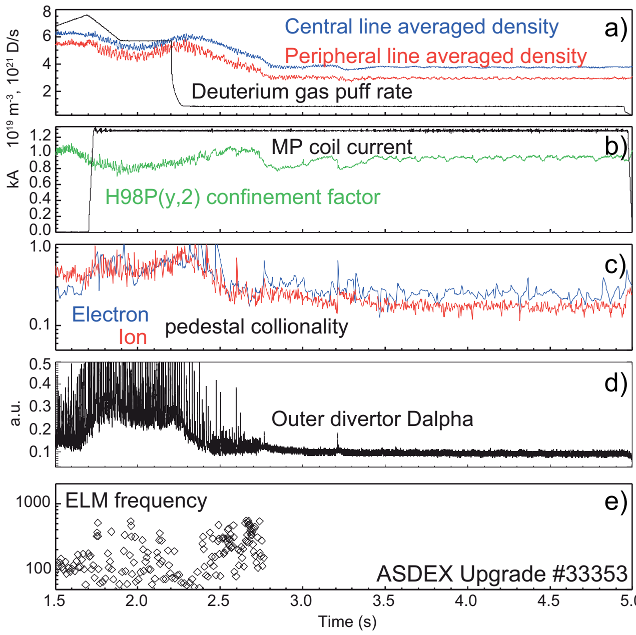

Figure 1: Typical time traces for full ELM suppression using n=2 MP.

ITPA endorsed AUG/DIII-D similarity leads to the discovery of full ELM suppression at low ITER relevant collisionality on AUG (see figure 1) with n=2 MP. The required high $\delta_{l/u}$=0.43/0.23 shape can only be accessed with ELM control during the low triangularity $\delta_{l/u}$=0.43/0.1 phase to avoid W accumulation due to a low ELM frequency. These discharges also maintain high confinement factor of $H_{H(98,y2)}$~0.95 despite the density pump-out compared to $H_{H(98,y2)}$~0.7-0.8 at low δ. ELM control with MP has also been shown to work in He discharges. For this extra effort has been undertaken to create clean He plasmas with high p_{ped} and large ELMs ($\frac{\Delta W_{ELM}}{\Delta W} \lesssim 15\%$). In this campaign for the first time NBI was operated in He and Ar frosting on the cryo-pumps was used to control the density on AUG.

ELM control was achieved at similar plasma densities as in D with an alignment of the perturbation transferred from the experience in D, demonstrating the transferability of the technique between species. ELMs are affected in a collisionality range where no ELM control is possible in D on AUG. ELM control via MP has also been studied in the ITER base-line scenario and its higher q_{95} alternative in both D and He. Whilst the correct alignment showed in both species a small density pump-out the ELM energy loss was not reduced. These scenarios exhibit small type-II like ELMs if shifted upwards or with decreased pumping speed. This is likely due to changes in the SOL density profile as discharges where gas fuelling was replaced by pellet fuelling demonstrate. With pellet fuelling during a type-II ELMy phase a lower SOL density develops at the same pedestal profiles and type-I ELMs reappear.

Transport and Confinement

The experiments under MST1 concentrated on the effect of impurities on the pedestal (see above), the effect of He ash on confinement, sawtooth driven impurity, momentum and particle confinement and JET identity experiments with respect to the ρ^* dependence of the intrinsic rotation. In particular the understanding of the latter is important for the low rotation plasmas expected in ITER and DEMO. Good identity matches with excellent data were achieved, but very different results in practically similar discharges question the commonly used analysis techniques.

Comparisons of the pedestal evolution in D, H and He with similar electron pedestal pressures before the ELM show in all cases similar phases of profile recovery and a mode that clamps the ∇T_e with the same mode structure. This suggests that the inter ELM transport is not affected by plasma species and ρ^*.

Disruptions and Runaway Electrons

EUROfusion dedicates a significant effort to disruption and runaway electron (RE) research, since they are two high priority issues to be solved to make fusion a success: in this framework key contributions came from AUG, which contributed to research on disruption prediction, avoidance and mitigation and on the suppression of high energy runaway electron beams.

With its long standing experience in the field, AUG provided new insights in the physics of disruption mitigation via Massive Gas Injection (MGI), a technique based on the injection of noble gas – either in the gaseous or frozen state – in the pre-disruptive plasma. AUG experiments have focused on exploring the influence of the amount of injected gas, and in particular on the search for the minimum quantity of gas for mitigation that is still compatible with ITER requirements in terms of radiation fraction and forces. The minimum amounts of gas used in the AUG MGI system has been decreased to $N_{inj} \approx 10^{20}$-10^{22} atoms, i.e. about two orders of magnitude smaller with respect to the maximum values used before.

The possibility of avoiding disruptions in high-risk plasma scenarios has been subject of experiments performed in AUG and TCV with localized injection of electron cyclotron waves on the MHD mode q=2 resonant surface, complemented also by the use of static or dynamic applied magnetic perturbations used to control the locked mode position or to entrain the mode.

The 2015/16 campaign in AUG (and in TCV) allowed the development, for the first time, of scenarios for the reproducible generation of RE during disruptions. Successful suppression of the runaway electron beam has been achieved with Argon injection 70 ms after the beam formation. The runaway electron current decay rate is observed to increase with the amount of injected argon. Suppression of the runaway electron beam has been achieved also with the application of an external magnetic perturbation produced with the B-coils and applied before the disruption takes place. When the poloidal spectrum of the applied n=1 perturbation is such that the normalized amplitude of the n=1 MP component dB resonant with q_{edge}=4 – the edge safety factor of the plasma just before the disruption – is maximum, significant beam suppression is observed.

Fast Ion Physics

Using the comprehensive array of fast ion diagnostics (FIDA, CTS) tomographic reconstructions of the fast ion distribution function, f_{FI}, are possible and can be directly compared to TRANSP modelling. This has been used to study the effect of sawteeth on f_{FI}. The observed changes agree well with modelling. Neutral beam current drive studies with on- and off-axis deposition in MHD quiescent discharges show that moderate anomalous diffusion D_{FI}=0.3 m^2/s is needed to explain the measured fast ion density profiles n_{FI} with on-axis NBI. With off-axis deposition n_{FI} is to flat to distinguish between reasonable levels of D_{FI}.

ECRH/ECCD was used to stabilize different Alfvénic instabilities. Studies of the effect of MP on fast-ion losses revealed a resonant transport process depending on the alignment of the perturbation in fair agreement with measurements.

Integrated Plasma Control

Significant progress was made in integrating disparate controllers into a coherent environment in AUG, with up to five plasma parameters controlled in parallel (core density, neutral pressure, β, divertor temperature and NTM stability). Actuator management, i.e., allocation of scarce resources in particular for ECRH, was demonstrated successfully. An unexpected coupling was observed between divertor-temperature and NTM control, to be addressed by future work.

Successful plasma position control was achieved by a multi-line-of-sight O-mode reflectometer, displaying remarkable resiliency even in the presence of strong ELM activity. When a high-field-side high-density front develops, the separatrix position is no longer estimated correctly but the scheme can still operate as a density-gap controller.

Control of Core Contamination and Dilution from W-PFC and Particle Throughput

AUG/MST experiments focused on identifying the dependence of tungsten accumulation in the core on MHD activity and external heating. Scanning the deposition location of ECRH outwards resulted in simultaneous occurrence of high- and low-frequency (1,1) MHD modes. The low-frequency mode appears to induce a hollow tungsten radial profile.
The particle throughput studies concentrated on H-mode conditions where the amount of cryo-pumping of the AUG vessel was varied. The main chamber pressure was directly correlated with local pumping, verifying the earlier observations in L-mode.

Detachment Control

Experiments were carried out to (a) study the dependence of detachment and the evolution of the X-point radiator on N, Ne, Ar, and Kr seeding at large P_{sep}/R, (b) investigate the effect of MP on detachment as well as on re-attachment, and (c) explore the influence of far SOL transport on the formation of high-density front at the high-field side. Neon was observed to lead to unstable core plasmas due to W transport inwards while the evolution of the X-point radiator could be tracked with all the seeding gases. In the future, the X-point radiator could be used in controlling full detachment. The magnetic perturbations, for their part, appeared to move the radiator front inside the confined region while at the divertor targets no noticeable effects were induced on the current and power flux profiles. The origin of the high-field side high-density front, for its part, was attributed to transport originating from the outer mid-plane.

Plasma Facing Components

Nitrogen migration was studied with the help of ^{15}N tracer injection from the outer divertor. The measured deposition in the main chamber was only 2-10 times smaller than at the divertor, indicating the strong contribution of the main-chamber walls on material migration. The generation of ammonia during N-seeded discharges and a considerable N and ammonia legacy on the AUG walls could also be observed. During the He campaign, erosion and surface modifications of W samples exposed to ELMy H-mode plasmas were studied. No net erosion was observed but all the samples had been covered with a thick co-deposited layer, potentially due to enhanced sputtering of W components in the main chamber. Close to the end of the campaign misaligned W samples were exposed to H-mode plasmas, and the results indicate flash melting by ELMs. The changes in SOL filamentary transport, in particular the formation of a density shoulder, were investigated in H-mode plasmas. Shoulders could be identified in inter-ELM phases provided that the effective collisionality $\Lambda > 1$ and D puffing was large enough.

Predictive Models for Divertor and SOL

The efforts focused on identifying ELM-induced heat loads on the divertor and main-chamber walls and to identify asymmetries between the inner and outer side of the vessel. Good data were obtained in different regions of the vessel and the analysis is ongoing.

3 Influence of the Density Profile Location on Pedestal Stability

Understanding the impact of main ion fuelling and impurity seeding on confinement is an important topic in fusion physics. Since present-day machines experience confinement reductions with the high gas puff required for stable operation with a metal wall, questions about how best to mitigate these effects have arisen. Nitrogen seeding in AUG and JET has been shown to increase confinement by up to 40%, but until now no mechanism has been found. The confinement changes have been localized to the edge pedestal, which has been the focus of recent AUG research.

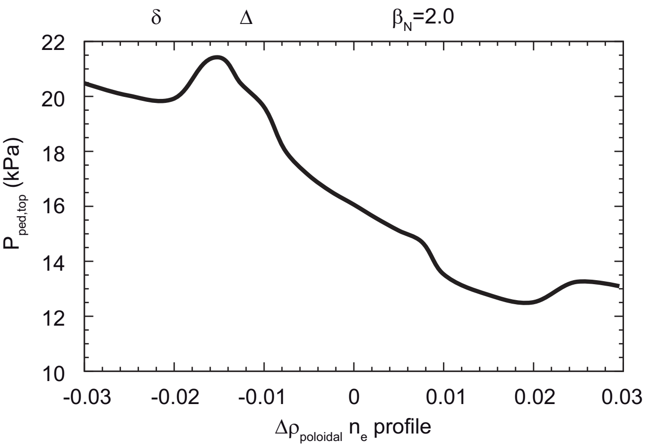

Figure 2: p_{ped} as a function of $\Delta\rho_{pol,ne}$ from the iPED code.

Given constant machine parameters (shape, I_p, B_t), several factors influence the edge pedestal top pressure (p_{ped}), such as the global β and the pedestal width. A critical parameter is the location of the pressure profile relative to the separatrix. Since the temperature profile is fixed by heat transport to ~100 eV at the separatrix, experiments focus on moving the density profile. A possibility to do this on AUG is offered by the high-field side high density (HFSHD), which has been observed on both AUG and JET. The HFSHD is a region of high density localised in the HFS scrape-off layer and is

formed by the ionization of recycled neutrals near to the HFS divertor entrance. Plasma is formed by power transported along field lines from the LFS to this region, while drifts play an important role in reproducing the observed experimental behaviour in SOLPS simulations. These simulations have shown that the pedestal density profile is expected to shift radially outwards when the HFSHD is present.
Modelling the effect of this shift using the predictive pedestal code iPED is shown in figure 2. The plasma parameters ($n_{e,ped}$, β_N, Z_{eff}) have been kept constant and the density profile has been shifted radially. A scan of the temperature pedestal top ($T_{e,ped}$) is then made and the critical p_{ped} is determined. With a shift of $\Delta\rho_{pol}$=0.01 (5 mm) p_{ped} is predicted to change by 25%, increasing with an inward shift and vice versa.
To experimentally test the impact of the HFSHD on the pedestal a gas scan was performed in a plasma at constant P_{heat}. The fuelling rate was increased from 0.5-2.0$\cdot 10^{22}$ e$^-$s^{-1}. Global stored energy decreased by 25% between these steps while the density in the HFSHD increased significantly.

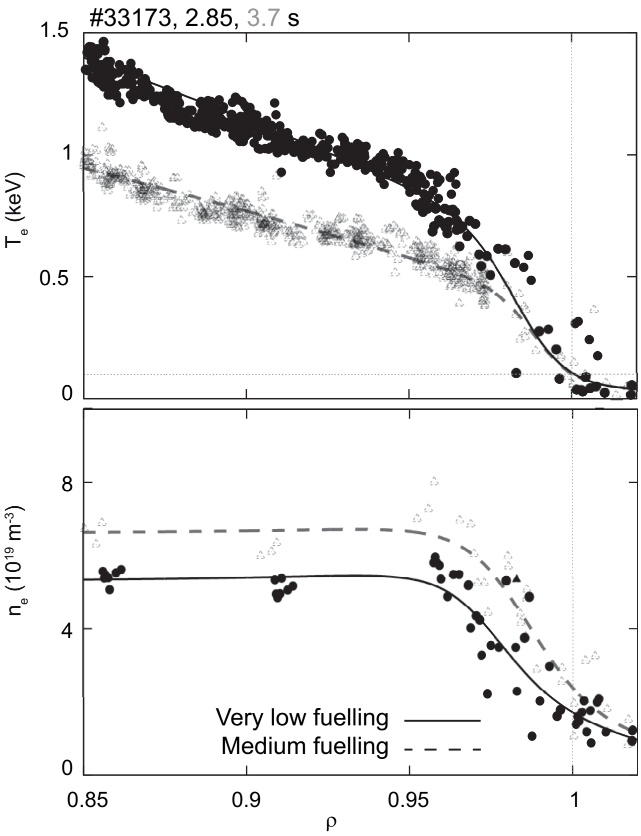

Figure 3: (a) T_e and (b) n_e profiles for low (black) and high (grey) fuelling.

Figure 3 shows the T_e (a) and n_e profiles (b) for the low (black, solid) and high fuelling (grey, dashed) phases. The temperature profiles have been aligned to 100 eV at the separatrix. Since Thomson scattering was used for both temperature and density, the density profiles are automatically aligned to the temperature profiles. Evident in figure 3(b) is a change in the position of the density profile; it has shifted outwards with $\Delta\rho_{pol}$=0.01. The $T_{e,ped}$ drop is not compensated by the increased $n_{e,ped}$, resulting in a p_{ped} loss of ~25%, echoing the prediction from the iPED code.
While the decrease of p_{ped} due to the formation of the HFSHD and the subsequent outward shift of the density profile does not bode well for future integrated scenarios in metal-walled devices, the density in the HFSHD can be reduced. Since plasma formation in the HFSHD relies on heat transported to the HFS divertor entrance, dissipating this heat by, for example, seeding nitrogen reduces the density in the HFSHD. Due to this reduction the density pedestal shifts radially inwards and p_{ped} increases.

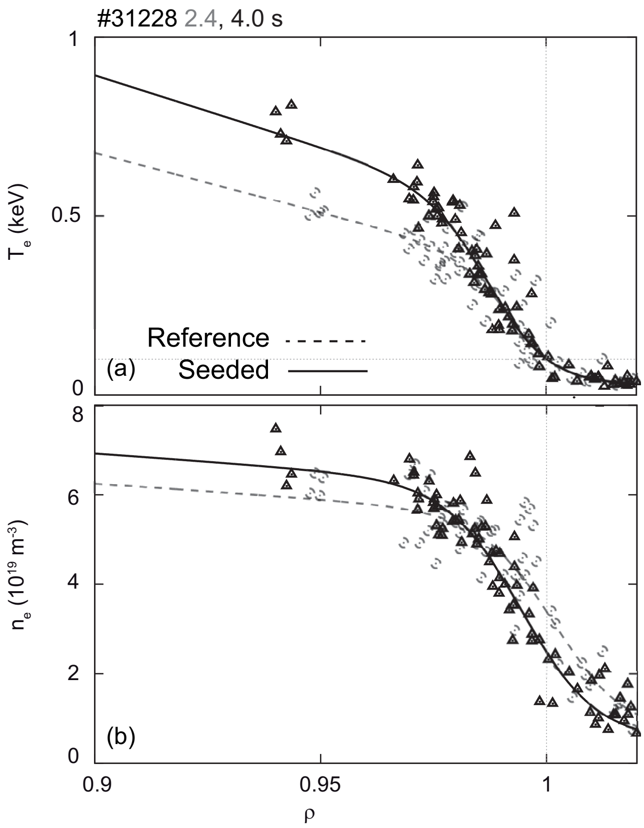

Figure 4: (a) T_e and (b) n_e profiles for reference (grey) and N_2 seeded (black) phases.

This is demonstrated in figure 4 for discharge #31228 with a reference phase at high fuelling (2.7$\cdot 10^{22}$ e$^-$s^{-1}, grey, dashed) and a nitrogen seeded phase (black, solid) where the density in the HFSHD has been reduced. $T_{e,ped}$ (a) has increased as a result of nitrogen seeding while $n_{e,ped}$ (b) remains similar, but the profile has shifted radially inwards. Linear stability analysis shows that both points are consistent with the peeling-ballooning theory, and have the same critical α_{max}.

However, the peak pressure gradient has shifted radially inwards to a lower value of q and, since $\alpha \propto q^2$, the same α_{max} corresponds to a much higher peak real space pressure gradient, leading to a higher p_{ped}.

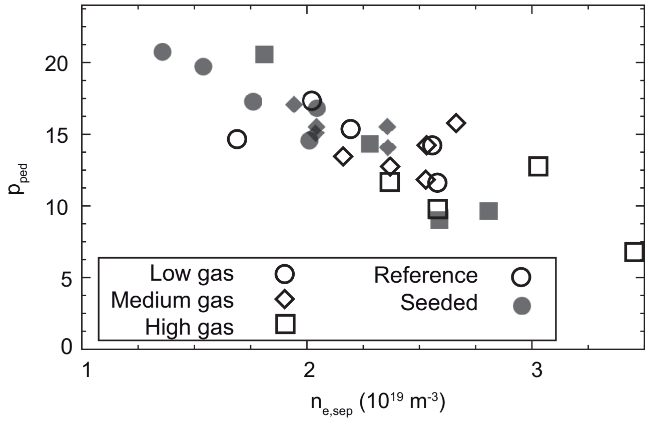

Figure 5: p_{ped} as a function of $n_{e,sep}$ from experiment.

To test this hypothesis over a wider range of conditions, a database was created consisting of a variation of gas fuelling, impurity seeding, and heating power. Predictive pedestal scans in density, global β, Z_{eff}, and the density profile location indicate that the density profile location is the dominant parameter determining p_{ped}. Since the location of the density pedestal is determined by the existence of the HFSHD and the diffusion of particles into the pedestal, the separatrix density $n_{e,sep}$ is a robust marker for the density profile location. Shown in figure 5 is p_{ped} (p_e+p_i) plotted as a function of $n_{e,sep}$. A strong correlation of p_{ped} with $n_{e,sep}$ over the variation in fuelling (1.0-2.0-2.7·10^{22} e^-s^{-1}), seeding (0-2% N_2 content), and heating power (5-15 MW) indicates that, for given machine parameters, this is indeed the dominating factor to determine pped and, hence, the global confinement.

4 The Role of E_r in Confinement at the Edge

The physics of the transition from the Low to the High confinement mode remains an important topic to be studied to improve predictions for future fusion devices. Both turbulent driven zonal flows and the neoclassical radial electric field are controversially discussed as possible key elements to generate sheared flows in the plasma edge which suppress turbulence and thus trigger an L→H mode transition. Earlier investigations on AUG showed that the radial electric field (E_r) profile in the edge of H-mode plasmas is consistent with neoclassical theory according to which it is, in the absence of strong toroidal flows, determined by the ion pressure gradient, $E_r = \nabla p_i / e n_i$. This result has recently been confirmed by CXRS measurements on the high and low-field sides of the torus in discharges with different ion masses (H and He).

Recent investigations related to the H-mode threshold power P_{thr} underlined the importance of the ion heat channel. The increase of P_{thr} with falling density could be attributed to a reduced collisional coupling between the heated electron and the ion fluid and thus a reduced ion heat flux (figure 6).

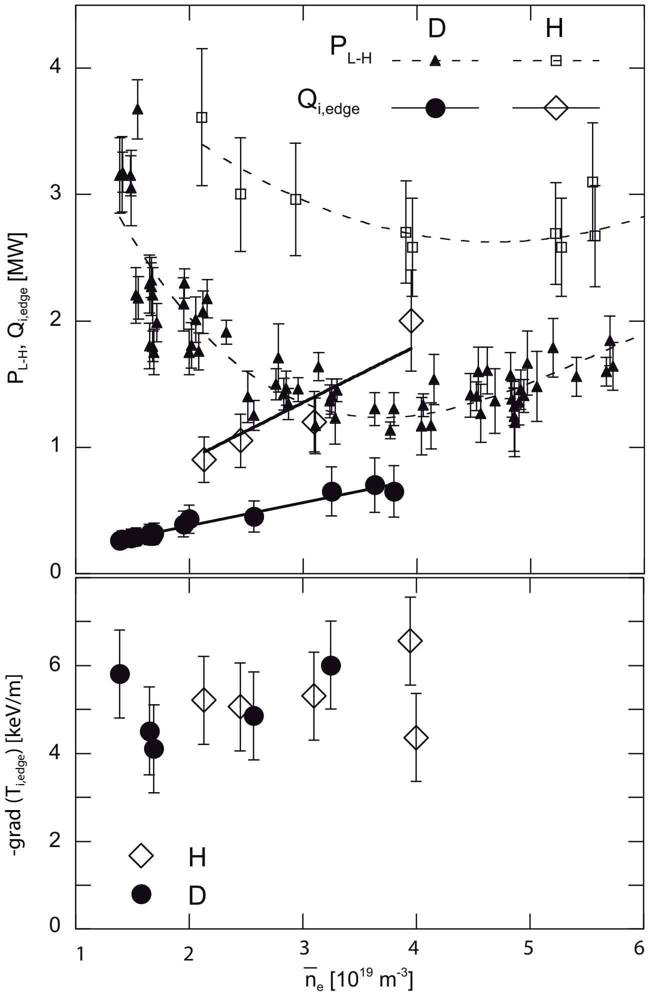

Figure 6: Power threshold, ion heat flux and T_i gradient at the plasma edge at the L→H transition.

In a similar way, figure 6 demonstrates that the isotope effect in P_{thr} (figure 6) results from the well-known isotope effect in the energy confinement time which leads to a lower edge ion pressure and neoclassical E_r in hydrogen as compared to deuterium. In both cases a higher heating power is required to establish the edge ion parameters needed for an L→H transition. This finding is important for the prediction of the density of the power threshold minimum for ITER.

According to figure 6 the L→H transition happens in all cases at a similar value for the edge ion temperature gradient and consequently same E_r. These data were taken at a toroidal field of B=2.5 T. The magnetic field scan in figure 7 demonstrates that the relevant parameter is not E_r both rather the E×B drift velocity and the related E×B shear flow. Figure 7 depicts the spectroscopically measured deepest value in the E_r well, which is proportional to the E_r shear flow, right at the L→H transition. This value increases linearly with the magnetic field strength B leading to the same flow velocity E_r/B. Therefore it seems to be the critical shear flow which needs to be reached to access the H-mode. Furthermore, a detailed comparison of edge profiles from recent discharges with a full-tungsten wall with earlier ones with a carbon wall revealed a steeper density gradient in the former, which is explained by a difference in the energy of recycled neutrals due to the heavier atoms of the wall material. A steeper density gradient demands only a shallower ion temperature profile in order to realize the required ion pressure gradient and E_r for the L→H transition. This explains why with a tungsten wall the H-mode can be achieved with about 30% less heating power.

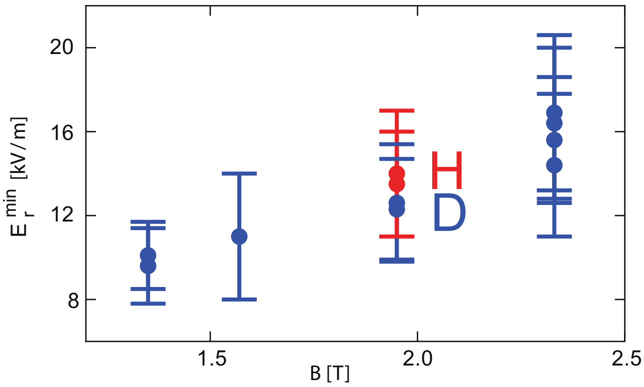

Figure 7: E_r-well minimum at the L→H transition for different B_t and isotopes.

In order to assess directly a possible role of a zonal flow which might contribute to turbulence suppression, high-resolution CXRS measurements were carried out during dynamic phases related to confinement transitions at the plasma edge with a time resolution of up to 0.1 ms. A clear indicator for the existence of zonal flows is a deviation of the measured E_r from the neoclassical prediction. From other fusion experiments such deviations are reported during the so-called I-phase, which occurs prior to the L→H transition and is often characterized as limit cycle oscillations between two confinement states. Figure 8 shows the detailed dynamics of a few I-phase cycles in the ion temperature and the poloidal impurity flow at three radii just inside the separatrix and in the density fluctuation amplitude form Doppler reflectometry at the plasma edge. Since the ion temperature and its gradient, seen in the distance between the time traces, and the poloidal flows of helium, as a measure of the neoclassical E×B flow, change in phase, a relevant contribution from a zonal flow, which would introduce a phase shift of $\pi/4$, can be ruled out. In variance to earlier results, the fluctuations are strong when E_r and the temperature gradient are weak. A direct comparison of CXRS measurements with neoclassical estimates was achieved by conditional averaging 30 I-phase cycles. The results are consistent with those from figure 8, the experimental and theoretical values agree very well and change in phase. Hence, unlike the observations from other devices, zonal flows do not seem to be important for the I-phase and the L→H transition, while effects on timescales faster than 0.1 ms remain unobserved. A systematic study of a large number of H-mode discharges revealed that each L→H transition is proceeded by a – albeit sometimes short – I-phase. The magnetic signals and the fact that precursors can be observed prior to I-phase bursts are reminiscent of type-III ELMs. They exhibit a poloidal up-down asymmetry which is explained by Pfirsch-Schlüter-like relaxation currents caused by ballooned transport losses. The strongest confinement improvement is observed at the transition from L-mode to the I-phase and not at the final transition from the I-phase to the H-mode which points to a limit cycle where MHD is more relevant than turbulence flow interaction.

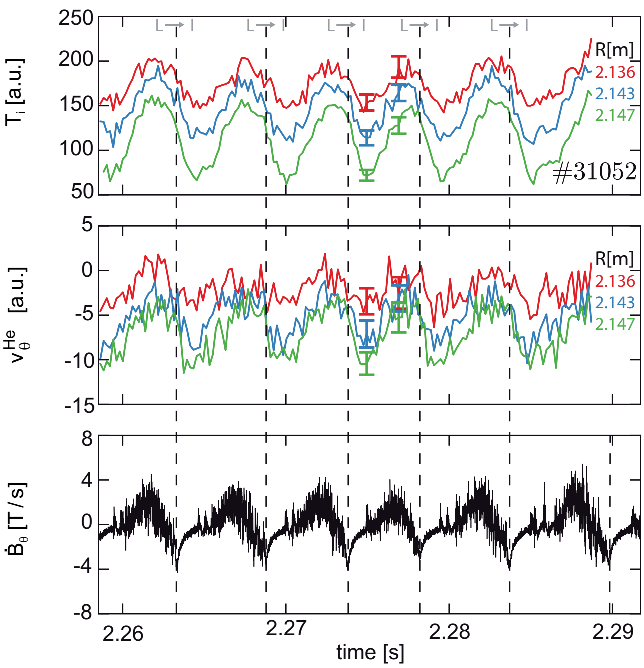

Figure 8: T_i, poloidal rotation of He and turbulence levels during an I-phase at the L→H transition.

5 j-profile Determination and Non-inductive Operation

The plasma current profile j and, closely related, the safety factor q occur in several important expressions of neo-classical and turbulent transport, such as bootstrap-current and critical-gradient-lengths. After the W-coverage of the inner surfaces of AUG, j-profiles based on the Motional Stark Effect diagnostic (MSE) were considered unreliable. Recently the issue has been tackled by implementing polarimetry and imaging MSE systems together with a thorough analysis of the existing MSE.

j and q only make sense in the framework of equilibrium reconstruction, therefore the latter was included into integrated data analysis, which directly allows a simultaneous interpretation of all available (partly redundant) information from the diagnostic improvements. In parallel, H-modes with q_{min} well above 1 were studied for the first time in a fully W-coated device using NBCD and ECCD resulting in almost non-inductive operation of the tokamak at q_{95}=5.3.

5.1 Magnetic Equilibria Combining Measurements and j-diffusion Modelling

The omnipresent ill-posedness of tokamak equilibrium reconstructions is mitigated by an integrated approach (IDE) of combining a comprehensive set of internal and external measurements with a predictive model of the expected current distribution. The goals using a sufficiently informative set of measurements and physical modelling are to have an overdetermined equilibrium, to overcome the need for non-physical regularisation (smoothing) of the source profiles entering the Grad-Shafranov equation, and to validate heterogeneous measurements due to partially redundant information. The set of measurements comprise magnetic probes, MSE, divertor tile currents, pressure profiles consisting of electron and ion thermal and fast ion pressure, loop voltage and isoflux constraints from electron and ion temperature profiles, which are recently extended by imaging MSE, polarimetry and a diamagnetic flux loop measurement. These measurements are complemented by flux-surface-averaged toroidal current distributions obtained by solving the current diffusion equation between successive equilibria. Additional current redistribution by sawtooth crashes identified using soft X-ray data is included by a Kadomtsev model or by a recent variant preserving the q=1 surface.

5.2 MSE Diagnostic

AUG's motional stark effect (MSE) diagnostic has been commissioned in the late '90s. While it performed well with vessel walls mostly carbon covered, a non-negligible systematic change of its field angle measurements with increasing tungsten coverage triggered a thorough analysis of its possible limits. Inspired by findings from the C-Mod tokamak, focused data mining revealed a systematic influence from broadband polarised background signal generated by reflections at vessel components into the view of the MSE diagnostic. Plotting the sum of two polarisation angles known to add up to 90 degrees (even in absence of an absolute calibration) versus plasma density and versus divertor neutral density made this evident and defined an operational range at low density where the data is not affected beyond normal measurement uncertainty. Additionally an absolute calibration of the MSE system down to approximately 0.5° remaining uncertainty could be performed for the first time by a combination of the accurate 3D space measurement system FARO used for spatial calibrations in the vessel with a newly built calibration light source, which can produce near 100% linearly polarised light using reflection at the Brewster angle between source and line of sight. Plasmas with higher density will be diagnosed correctly with the emerging polychromator system which simultaneously measures the polarised background. Provided the adequate choice of NBI sources, the MSE system is then expected to deliver reliable internal magnetic constraints for equilibrium reconstructions also in the full-W device.

5.3 Imaging MSE

The imaging MSE (IMSE) approach utilises the same Stark-split D_α light emitted by the neutral beam particles as conventional MSE systems. However, this light is led through a series of birefringent plates, a polarizer and imaged onto a CCD chip, forming an interference pattern. While conventional MSE systems typically filter out the σ- or π-lines of the Stark spectrum, IMSE diagnostics utilize all of the lines, which increases the signal to noise ratio and eliminates the need of using precisely tuned narrow band filters. Furthermore, they are not disturbed by polarized broadband light, like reflections from the metal walls. IMSE diagnostics provide a 2-D image of the polarisation angle, which significantly increases the quality of the equilibrium reconstruction compared to 1D systems. A new permanent IMSE diagnostic has been in operation since the 2015-2016 experimental campaign.

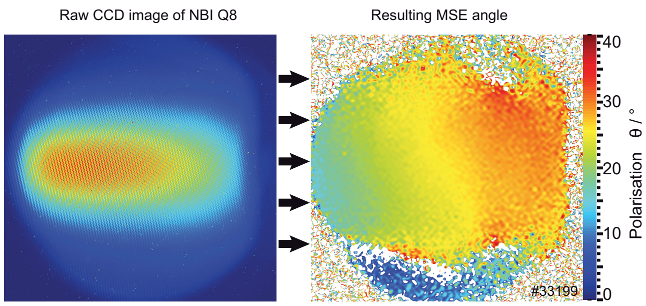

Figure 9: Raw IMSE image (left) and demodulated polarisation angle image (right) for a single time point.

The system has a wide field of view imaging from the outer separatrix across the magnetic axis and up to 10 cm on the high field side. The optics of the AUG IMSE system are designed for very low Faraday rotation, which is monitored, together with possible drifts, using in-vessel polarized light sources. For the 2016 experimental campaign a prototype 'back-end', the set of lenses and crystals designed to create the interference pattern on the CCD camera, was mounted to the new optical relay system. It was possible to resolve changes of the polarization angle of 0.1° at a time resolution of 5.6 ms, enabling the study of plasma current redistribution during sawteeth. For selected discharges, the IMSE data was successfully integrated in the equilibrium solver producing excellent results.

The prototype back-end will be replaced by a fully optimized system before the start of the 2017 experimental campaign. The new design features larger birefringent plates yielding a larger étendue, higher stability and improved calibration possibilities.

5.4 Polarimetry

AUG used to have a pure interferometer until recently, which only registered the phase shift experienced by the probing beams (λ=195 μm), but not the rotation of polarisation due to the Faraday effect, which is given by

$$\alpha = \frac{\lambda^2 e^3}{8\pi^2 \varepsilon_0 m_e^2 c^3} \int n_e(s)\, B_{\parallel}(s)\, ds$$

For the given poloidal geometry, $B_{\parallel}$ results from the poloidal field, which is caused by the plasma current. In the 2015/16 experimental campaign, two interferometer channels were equipped with an additional polarimeter with an angular resolution of about 0.6°.

The measurement is routinely available throughout the entire discharge and for all heating schemes. Biasing factors, such as polarized ECRH stray radiation and the pick-up of a toroidal field component due to a slight tilt of the viewing chord out of the poloidal plane, were identified and countermeasures were developed. Figure 10 illustrates the response of the measured Faraday angle as plasma current is expelled from the plasma center. Reduced central current means reduced $B_{\parallel}$ on the inner viewing chords, and therefore reduced $n_e \times B_{\parallel}$, such that the Faraday angle drops in relation to the line-integrated density.

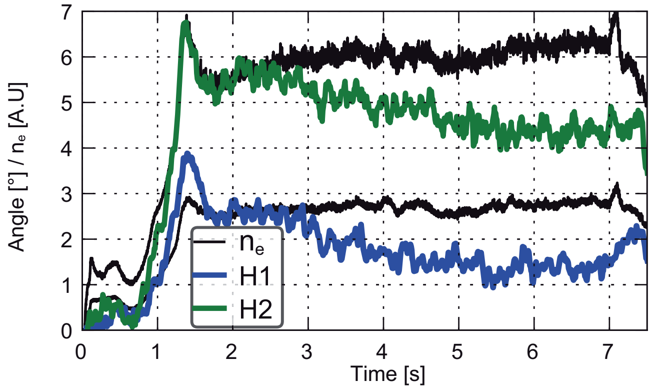

Figure 10: Temporal evolution of the Faraday angle (colour) and the corresponding line-integrated density (black) for two lines of sight. Reduced central current results in a strong drop of the Faraday angle on the inner viewing chord H1, while the mid-radius chord H2 shows a smaller relative change.

5.5 Non-inductive Current Drive

The approach to achieve scenarios with high non-inductive current fraction in AUG is based on simultaneously maximising neutral beam current drive (NBCD) and bootstrap current (BS). The necessary high pressure gradient for the latter is achieved by high heating power (12.5 MW NBI, 2.5 MW ECRH). The feedback controlled neutral beam power (lower part of figure 11) produces a pre-programmed β-ramp to slowly approach to the stability limit. The preferential use of the more tangential off-axis neutral beams maximizes the off-axis NBCD. Additionally all available ECRH gyrotrons are used for off-axis ECCD to form a hollow current profile thus enhancing the bootstrap current.

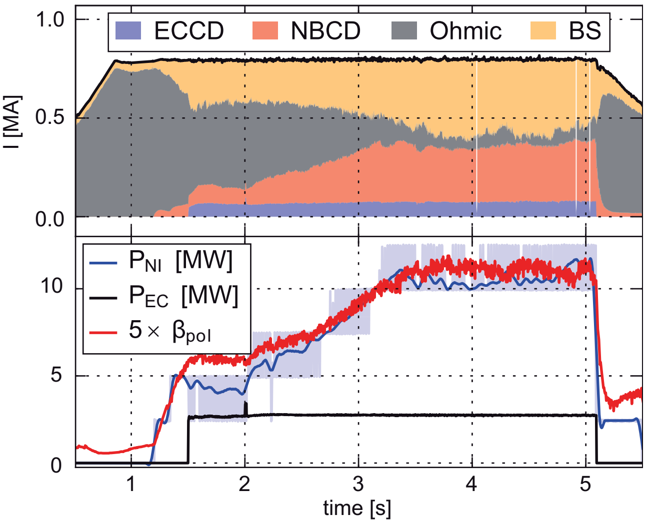

Figure 11: Plasma current composition of pulse #32305.

The upper part of figure 11 shows the IDE-calculated current composition. The ohmic current, driven by central solenoid in plasma-current feedback control mode, is more and more replaced by NBCD and BS. In the end of the discharge the ohmic current fraction is less than 5% of the 0.8 MA plasma current, the rest being 45% NBCD, 40% BS and 10% ECCD. However, due to the long current diffusion time scales steady state is not fully reached.

6. Reduction of ICRF-specific W Production by 3-strap Antennas

The tungsten (W) production, specific to the Ion Cyclotron Range of Frequencies (ICRF) heating, and the resulting enhanced radiation had a strong influence on applicability of ICRF in the all-tungsten AUG. One of the main reasons for this increased impurity production is likely the RF induced electrical fields near the antenna, in particular the parallel field $E_{\parallel}$. A significant step to reduce these fields was the installation of the new 3-strap antennas [1] in 2015. Characterization of these antennas with W-coated limiters and a comparison to the 2-strap antennas with boron (B)-coated limiters is presented below.

The 3-strap antennas are equipped with local RF image current measurements. The local amplitudes of the total RF electric field and of $E_{\parallel}$ can be assumed to be, for a fixed limiter geometry and loading conditions, directly proportional to the RF current measurements and can be compared with the values computed by electromagnetic codes.

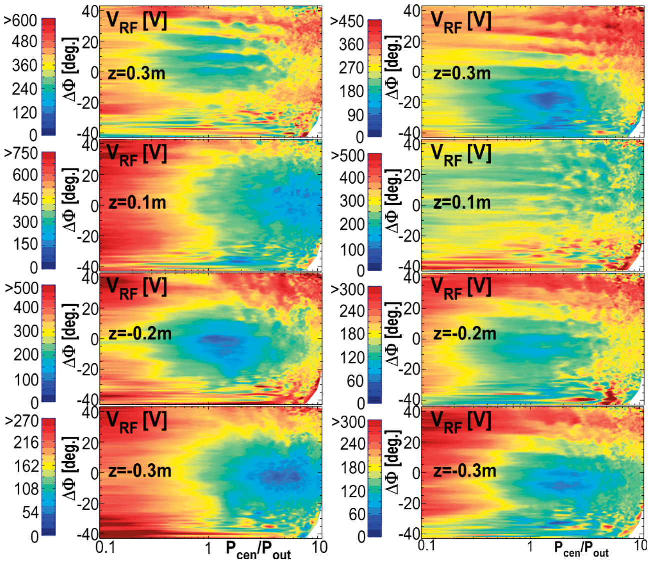

Figure 12: RF voltage V_{RF} measure, as a function of fraction of RF power to the central stra and phase difference, at the limiters of a 3-strap antenna in sector 12. Left column: left limiter as seen from plasma. Right column: right limiter.

The response of the RF current measured at the limiters of the 3-strap antenna in sector 12 to a scan of the power balance between the central and the outer straps P_{cen}/P_{out} and of phase deviation from dipole ΔΦ is shown in figure 12, where we plot the RF voltage V_{RF}, equivalent for a 50 Ω load in vacuum estimated from the RF current measurements. The experimental data of figure 12 can be compared to calculations, with TOPICA, of the local $E_{\parallel}$-field averaged spatially over the corresponding locations of the limiters in figure 13.

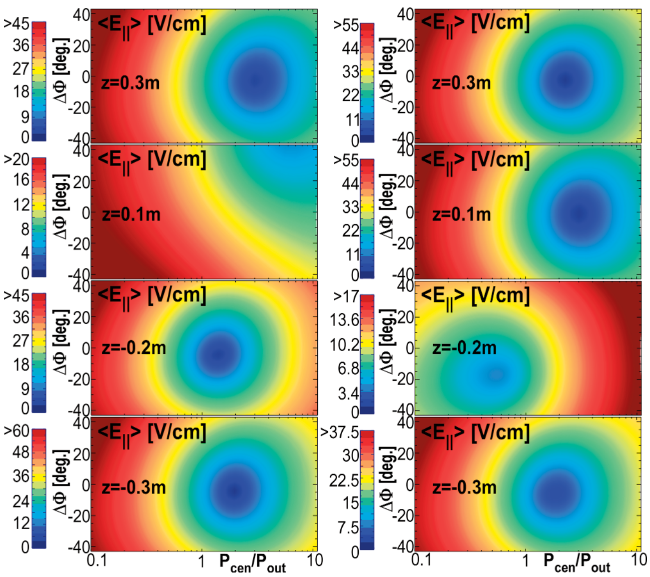

Figure 13: RF amplitudes as a function of fraction of RF power to the central strap and phase difference, calculated by TOPICA for a flat antenna model for the locations corresponding to figure 12.

The measurements and the numerical results agree qualitatively well on the existence of minima of the RF quantities and on the shifts of these minima with P_{cen}/P_{out}. The agreement of the exact values of P_{cen}/P_{out} and ΔΦ for the minima is less good in some locations close to vertical position z=0.1 m where the qualitative behaviour of the dependencies experiences a change. Nevertheless, in general the local RF measurements at the limiters are represented well by the locally excited RF field calculated by the TOPICA code [2] which does not take into account the slow wave propagation. This corroborates the principle of minimizing the RF image currents used for optimization in the design of the 3-strap antenna.
The W sputtering at the limiters presented in figure 14 shows a good correlation with the RF amplitude measurements figure 12 close to z=0.3 m and z=0.1 m. This confirms that reduction of the RF image currents at the antenna reduces the W sputtering at the antenna.

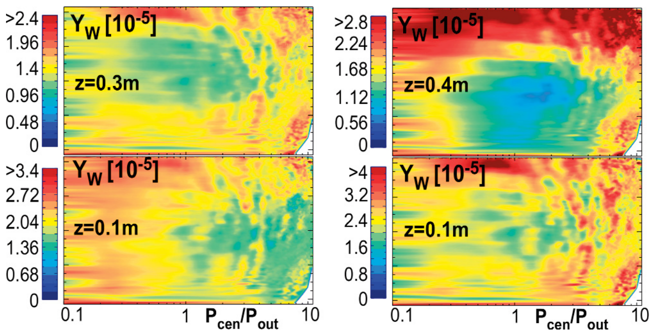

Figure 14: W effective sputtering yield Y_W in the locations close to the RF measurements. Left: left limiter as seen from plasma. Right: right limiter.

The effect of the antenna design on the rectified DC sheath potential $V_{DC\ sheath}$ has been estimated by the asymptotic version of the SSWICH-SW code [3] which uses the RF field maps from the RAPLICASOL code [4].

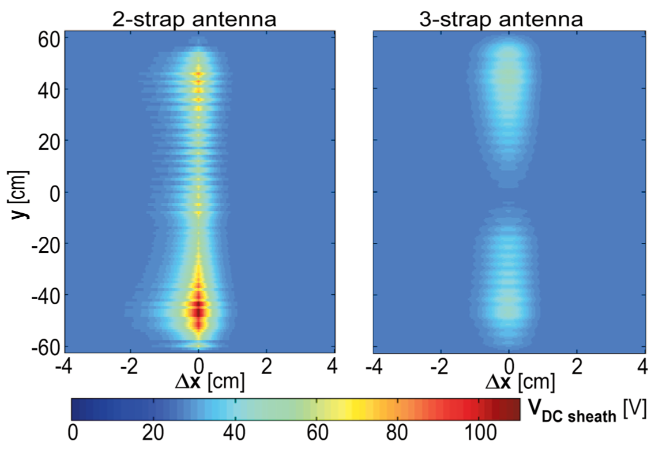

Figure 15: SSWICH-SW calculations of $V_{DC\ sheath}$ with flat models of 2-strap (left) and 3-strap (right) antennas for 1 MW per antenna.

Figure 15 presents the calculations of a radial-poloidal distribution of $V_{DC\ sheath}$ on the leading edge of the antenna side limiter using the radial distance from the leading edge Δx, both for the original 2-strap and for the 3-strap antenna with P_{cen}/P_{out}=2.0.

In the experiment, relative differences with respect to the B-coated antennas in figure 16 help to estimate the improvement of the W-coated 3-strap antennas compared to the W-coated 2-strap antennas. Prior to the installation of the 3-strap antennas, the use of the W-coated 2-strap antennas led to significantly higher W content in the plasma. At least a factor of 2 higher W concentration was found in comparison to the B-coated 2-strap antennas. A comparison between the 3-strap W-coated antennas and the 2-strap B-coated antennas is shown in figure 16. The 3-strap antennas operation results in about the same W source characterized by Y_W measured at the right limiter of the 3-strap antenna in sector 12 (spatially averaged over locations) and about the same core W content as that for the B-coated 2-strap antennas for a given ICRF power. Thus the improvement in the experiment corresponds approximately to the improvement expected from figure 15.

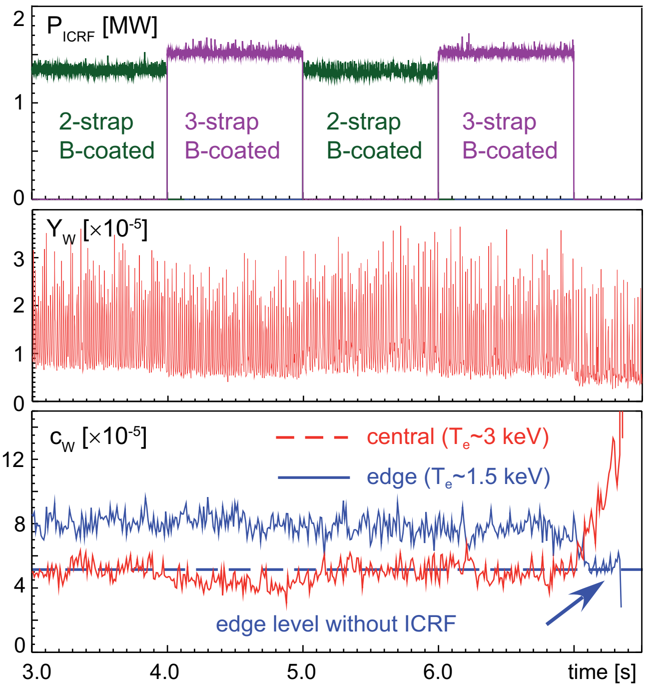

Figure 16: Comparison of Y_W measured at the right limiter of the 3-strap antenna in sector 12 and W concentration in the H-mode plasma during operation of the B-coated 2-strap and the W-coated 3-strap antennas.

7 Technical Systems

The experimental campaign 2015/16 was started in May 2015. It was the second campaign with the solid tungsten divertor Div-III. The experimental campaign comprises in total 1980 pulses with 1540 pulses useful for the physics programme. 466 discharges were heated with more than 10 MW, 137 of them with more than 15 MW. A maximum heating power of 25 MW was applied in 4 pulses. The experimental campaign was finished with discharge #33724 on May 19th 2016.

7.1 Machine Core

The timeline of enhancement and operation is summarised in figure 17.

The shut down 2014/15 was used to install a pair of new 3-strap ICRH antennae reducing the electrical fields in front of the antennae as reported below, to modify the support structure of the lower outer divertor and to prepare the inner column for P92/Eurofer installation. During operation in 2013 divertor modules were misaligned due to strong eddy currents induced during disruptions resulting in leading edges in between sections. The eddy currents are avoided by isolating one of the two fixing points of the divertor substructure at the vessel. In addition detailed FEM calculations have shown that the few broken tungsten tiles can be explained by an eddy current induced in the cooling plate bypassed by the tungsten target due to the lower resistivity of tungsten compared to stainless steel. This by pass current and the resulting force is reduced by applying the same solution for isolating the clamping at the top of the target as used for shunts. These measures have significantly reduced the forces acting on the solid tungsten divertor which could be operated without limitations during the 2015/16 campaign.

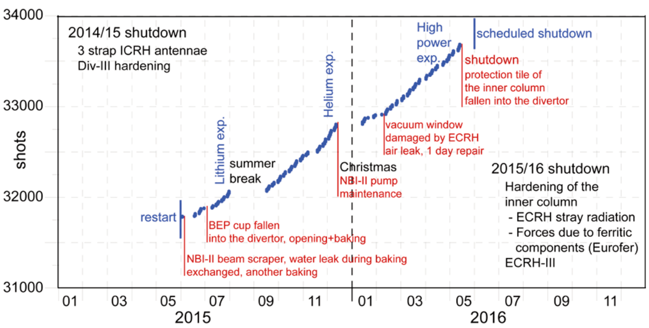

Figure 17: Time line of AUG enhancement and operation in 2015/16.

The inner column was operated with two rows of ferritic steel parts (P92/Eurofer) as contribution to DEMO. These tiles were designed with labyrinth like structure to reduce ECRH radiation behind the inner column. A few edges of graphite tiles were broken in the region overlapping with P92. Detailed analysis of the structure of the inner column by FEM calculations has revealed that this is due to a tilting of the inner column support structure during disruptions. The 1991 design includes a large loop (about 1 m^2) receiving dB/dt during disruptions. This loop will be broken by insulation the two subcomponents without losing mechanical stiffness during the 2016 shutdown. In addition the internal stiffness of the inner column support structure is increased by welded inserts.

This allows installing P92 in the central part (row 4-10) of the inner column.
In addition a huge set of diagnostics were refurbished or newly installed.
During the operational campaign we had two unscheduled openings followed by baking and one short venting. During the baking in preparation of the restart a water leakage in the NBI-II dump (S7) appears. The beam dump was taken out for repair. It was replaced by the beam dump structure used until 2007. About 100 discharges after restart a collimator tube of an optical diagnostic (BEP) was falling down into the lower divertor and jumps around the torus during discharges. AUG was shut down to remove the tile, to reinstall a new collimator tube and to improve the fixing. In February 2016, during discharge #33017, ECRH stray radiation was absorbed in a vacuum window that became leak. It could be replaced from outside by venting the torus with nitrogen for about 1 h. Plasma operation could be restarted w/o baking. Two weeks before the scheduled shut down after a disruption at the end of #33689 a protection tile of the inner column was falling down into the lower outer divertor and the vessel was opened after discharge #33724.
During the campaign 2015/16 a modified IR-system measuring in the 5 μm region was used for in vessel viewing in between discharges. The sensitivity of the IR-system allows receiving in vessel pictures without external illumination in contrast to the video protection system that relies on plasma radiation for illumination.
In June a seven month shutdown was started for hardening the high filed in-vessel bellows (isolation bellows) against ECRH stray radiation and to adapt the support structure of the inner column to cope with the increased forces due to Eurofer like ferritic steel as plasma facing component.

7.2 Gas Inlet and Torus Pumping System

Up to now the maximal hydrogen inventory in AUG is restricted by safety limits of the roughing pumps to 3 bar·l. To allow a higher gas input, as required for example at high radiation scenarios, enhancements of the pumping and gas inlet systems had been implemented. Most of the gas inlet is stored at the in-vessel cryo pump and released during regeneration within some seconds leading to a pressure rise, which influences diagnostics and loads up the wall again. For operation a full day of plasma discharges without warming up the LHe panel of the cryo pump would be desirable. This requires a new safety limit of about 200 bar·l of hydrogen, which will be stored at the cryo pump and released during regeneration. To keep the performance of the existing pumping systems and to minimize changes, a new dry screw pump certificated for hydrogen operation (Leybold DV 650) is directly connected to the vessel. For regeneration the torus vacuum will be isolated from all other vacuum systems and the cryo pump will release its inventory. A pressure up to 5 mbar is expected, which will be reduced by the hydrogen certified pump to 0.01 mbar, which can be handled easily by the existing turbo pumps.
The operation of systems working with explosive gases has to be permitted by the safety authority. For this purpose a survey report in cooperation with the DEKRA, who has the competence to evaluate explosive systems, was performed. This required an inventory taking of the existing and new systems and a documentation using process charts. The proposed new installation fulfills all requirements and will be established during the 2016 maintenance phase. To enable operation with a higher inventory failsafe communication with the gate valve control and the cryo pump is established
Up to now the gas inlet is determined by the flux of the individual valves and integrated by the control systems, resulting in the upper limit for the hydrogen inventory of the cryo pump. For the operation using bigger amounts of hydrogen a more failure tolerable system is required. An extra volume, containing the maximal allowed amount of hydrogen, is inserted in between the gas cylinders and the switching matrix, which supports the piezo valves. The pressure inside this temperature monitored systems allows determination of the gas inlet, i.e. the maximal possible inventory on the cryo pumps. As a refill is only possible after an regeneration, the hydrogen inventory is physical limited.
Gas inlet into the torus is realised using piezo valves with internal flow measurements. As the system was installed before the first startup 1991, a renewal of components was started in the last years. After optimising the gas feed, signal storage and electrical connections and overhaul of the valve itself has now started. A replacement of the sealing and the piezo crystal requires a new mechanical adjustment and recalibration of the control electronics. For this purpose a new device was set up, which allows to measure the leak rate and flux of the valve. Activities to enhance the maximal throughput of the individual valve were also started.
To allow a versatile gas inlet using different locations and kinds of gas, 19 piezo valves feeding the plasma direct or using tubes are actually in operation. To take the conductance of the different tubes into account the valves have been calibrated in situ for several gas species.

7.3 Experimental Power Supply

From the technical point of view, 2015 and the first half of 2016 was dominated by the final installation and commissioning of the BUSSARD inverter system and a long operating period. For details of the BUSSARD system see figure 18.
A cultural highlight was a theme concert organised by the Bayerische Staatsoper in cooperation with the Max Planck society. The generator hall of flywheel generator EZ4 – associated with excessive noise during operation of the experiments – was filled with classical sound by some members of the Bayerische Staatsoper. Another big attraction was the

Long Night of Science during the celebration of the 1100^{th} anniversary of Garching. This time, generator EZ2 was illuminated with mystic light and the crowds of interested visitors did not come off until midnight.

The second MST1 campaign was very successful from the Experimental Power Supplies (ESV) point of view. There were only few problems affecting the AUG physics program. Load-shedding after pulse-stops resulting in overvoltage trips of generator EZ3 and converter trips because of voltage unbalance by inaccurate switching of reactive power compensation modules represented the most serious problems. Most exciting and successful was the operation of the first four BUSSARD inverter modules operating on a set of 4×4 in-vessel B-coils. After two years of mounting and commissioning, at the end of the 2016 experimental campaign all 16 inverters were handed over for physical use. Every B-coil can now individually be operated with current up to 1.3 kA and maximum frequency up to 500 Hz. After transfer of two 15 Mvar reactive power compensation (RPC) modules from generator EZ3 to EZ4, the RPC of EZ3 was rearranged. Instead of 6×15 Mvar modules, EZ3 now operates with 2×15 Mvar and 2×30 Mvar modules. The vacant switchboards will be used for the extension of the RPC of EZ4. All required components are already on site. Installation and commissioning is scheduled for the 2016 AUG shutdown.

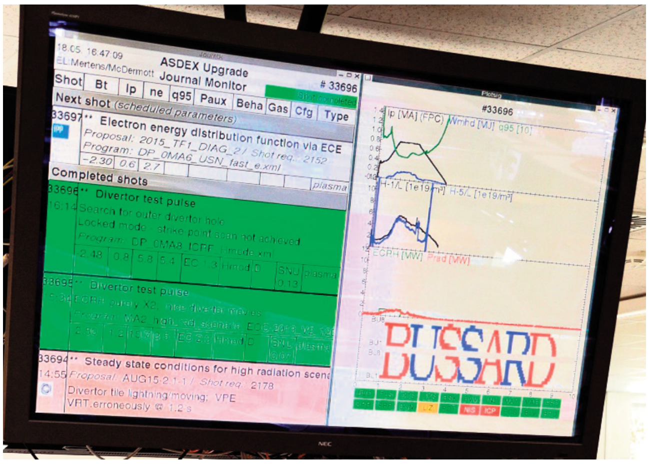

Figure 18: The BUSSARD logo shown on the AUG journal monitor was generated by the help of real coil currents during plasma pulse #33696 to demonstrate full operation. Each row of the logo represents a coil current – for 16 individual coils with separated inverters. The X-axis represents the variation of currents in time. Each red pixel stands for a coil current of -1 kA, each blue one for +1 kA. The full duration was 10 s. Unfortunately the plasma had to be ramped down after 1.6 s due to a locked mode safety message generated by the 'B' in BUSSARD.

Generator EZ4 received a new sound protective hood and the new cooling tower to be installed in 2016 has been delivered. The installation of the first high voltage power supply system for ECRH-3 has been completed and the pre-commissioning on a dummy load started. Testing on the ECRH-3 system should go on after installation of the first gyrotron. During the upcoming shutdown, numerous maintenances and modifications of the installation are scheduled. The inspection of generator EZ2 already revealed some defects that absolutely justify the narrow service intervals also for the future.

7.4 Neutral Beam Injection

Overall neutral beam injection (NBI) worked reliably throughout the 2015/16 campaign. As in previous years local heating of the beam duct due to re-ionised beam particles was a general problem in high power pulses. A video diagnostic installed to prevent melt damage failed early in the campaign and could not be repaired without venting. Thus deprived of early warning melt damage occurred again, however without implications for the ongoing campaign. A new box exit scraper and an improved version of the video system are expected to finally solve the problems. A leak in the calorimeter of box 2 occurred in December 2015. With the calorimeter drained and defunct operation was nevertheless possible until the end of the campaign due to the high reliability of the rf beam sources that required no conditioning pulses.

In 2015 helium neutral beams were injected in a total of 44 discharges during the He campaign in December. AUGs NBI is primarily designed for deuterium injection and delivers 20 MW from two injectors with four beams each at 60 and 93 keV, respectively. The main obstacle for helium operation is that the Ti getter pumps ($3 \cdot 10^6$ l/s for hydrogen) do not pump helium at all, leaving only the conventional pumping system with $<6 \cdot 10^3$ l/s.

Serious trials to operate the AUG NBI with He had already begun in 2014. It was found that despite the lack of high speed pumping up to two beams per injector could be operated simultaneously at reduced feed gas flow without particular restrictions on the beam-on time. Achieving a tolerable stationary pressure in the beamline in long pulses relies on the effective pumping of the NBI box through the beam duct to the torus. As a consequence the neutral gas flow into the torus is the dominant source of He in these discharges.

For injector 1 the power per beam was limited to 0.55 MW at 40 keV by the required filament current in its arc sources, while for injector 2 the limitation came from the bending magnet's power supply that restricted the beam energy to 68 keV and the NBI power to 0.73 MW per beam. Thus the maximum NBI heating power in He is 2.6 MW for a full plasma duration of up to 10 s.

7.5 Ion Cyclotron Range of Frequency

Two new antennas, targeted to reduce impurity generation, were designed and fabricated at different sites in an international cooperation between IPP, ASIPP (Hefei, China) and

ENEA (Frascati, Italy). In one antenna, a multi channel reflectometer system, designed and built by ENEA and IST (Portugal) is integrated. Following the removal of previous antennas at the end of 2014, installation of the new antennas, supported by staff of ASIPP, ENEA and IST started in the middle of February 2015. To use the available in-vessel time to the best possible extent, a staff of two was allocated inside the vessel for each antenna, while on the outside a staff of two ensured the flow of materials and tools. Despite the complexity of the work, partially in a very congested environment, the installation of both antennas, one of them including reflectometry, was completed in just eight weeks.
The AUG campaign re-started in May 2015 and both antennas had been commissioned by mid June 2015. The new antennas show a significant reduction of W impurity production, consistent with predictions. Results of the new multi channel reflectometer system correlate well with other diagnostics where comparisons are possible, and it provides in particular density profiles with poloidal and toroidal resolution in front of the antenna. This unique tool will help to analyze local effects in front of the antenna (such as convective cells) and support coupling studies.

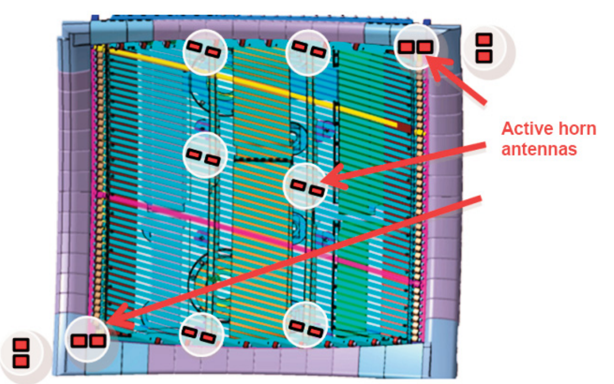

Figure 19: Positions of the ten reflectometry antenna pairs, presently three-pairs are activated and fully instrumented.

Tests during the last campaign have confirmed that the W-impurity generation is at a minimum when the power ratio of central strap to outer straps is about two. Since presently the installed power for the central strap and the outer strap is both 1 MW per antenna, half of the installed power on the outer strap cannot be used in the optimal configuration. To improve this situation, the installed power to the central straps will be doubled. An additional 2 MW RF generator (in the past used by the ASDEX and Wendelstein 7-AS experiments and recently modified to use a modern tetrode) is being integrated into AUG's ICRF system for the 2017/2018 campaign, to provide additional power to the central straps of two antennas, resulting in a total of 10 MW ICRF installed generator power.

7.6 Electron Cyclotron Resonance Heating

In the 2015/2016 campaign the ECRH system was operating with 8 gyrotrons until end of 2015, when the old ECRH-1 system was closed down to prepare the modifications of its lauchers and the replacement of the torus-hall cooling system and cabling for ECRH-3 in the shutdown (see below). Operation of ECRH-2 was reliable, though operation at full power still requires conditioning of the transmission lines. The enlargement of the coupling mirror to the torus for unit 4 (design IGVPT, Uni Stuttgart) successfully reduced arcing in this area. ECRH-1 power was limited by developing side lobes in two 20 years old gyrotrons (inherited from W7-AS).
The ECRH-3 project replacing the ECRH-1 system by a 4 MW, 10 s system achieved completion and testing of major subsystems including the cooling and the control system. The transmission system from the matching optic units to the top of the torus is ready and aligned including a new long pulse load based on turbulent water circulation in teflon tubes (design and assembly IGVPT). The load was successfully tested by the ECRH group of W7-X. The electronic subsystems from IPP workshop (ITZ) were delivered and tested or are still on track. The first body modulator is at the factory acceptance test, it achieves the design parameters, but occasional failures still need to be resolved. The first series modulator with a new tetrode from Thales has been built by ITZ and has been tested successfully. Series-modulators 2 and 3 are transferred from ECRH-1. The last series modulator will use another new Tetrode and a new socket which are expected still in 2016. ECRH-3 will have its own HV-modules based on old modules from DESY. First tests with the series modulators on an ohmic load allowed longer pulses but also showed that the digital control of these HV-modules still needs further development to cope with on-off modulation. The acceptance test of the first gyrotron is planned for October 2016. Delivery of the other gyrotrons is expexted in autumn 2016 as well as spring and autumn 2017. The 2016 shut down is used to modify the ECRH-1 launchers to be capable to handle the 10 times higher energy of the ECRH-3. This requires changing from BN to diamond windows implying a change of diameter from 120 to 87 mm. For all four launchers the B-port flanges were removed to modify the flanges for the EC windows by AUG workshops. Outside the torus all 120 mm transmission components will be replaced. Inside the torus, solutions vary for Sectors 6 and 8 (S6,8) holding one launcher each and sector 14 (S14) which holds two launchers. For S6,8 the launcher mirrors are large enough to cope with the larger divergence of the 87 mm beam, only the fixed stainless mirror is replaced by a Cu coated graphite mirror in order to reduce ohmic losses and mirror heating. In S14 the 120 mm beams were guided through the B-port with waveguides, which close to their end on the plasma side supported the fixed mirrors. In order to minimize changes this wave-

guide is replaced by a mode converter of same outer dimensions which transforms the HE_{11} mode with 87 mm diameter to a Gaussian mode of 120 mm, to be handled by the old optical system (design and fabrication IGVPT). Replacement of the 120 mm waveguides and readjustment of the launchers is the major part of in vessel work for ECRH-3 in 2016. In order to accelerate the mirror movement the bearings of the gearing mechanics of the launcher in S8 have been redesigned by the Institute of Machine Elements from the Technical University of Munich. After production at ITZ, assembly and testing in a vacuum oven, the gearing block of the launcher in S8 is planned to be exchanged in the summer opening 2017. If the concept is successful the other launchers will follow. We aim for a real-time control of the launching angle of a few degrees per second especially for current drive control in advanced scenarios.

7.7 CODAC

The Discharge Control System (DCS) started the campaign with a major upgrade to the parameter server, GUI and the interface to diagnostics, as well as a better organised code structure. Some initial problems could be quickly resolved so that DCS regained its customary reliability, providing a stable basis for a wide range of physics experiments. The upgrades also improved the portability of DCS, enabling its use as the framework for the WEST control system. Final preparations are being made for the first WEST plasma in the coming months
One of the foci for the AUG campaign was on integrated control. Dedicated experiments controlled all of β, NTM, core n_e, neutral particle pressure and divertor temperature simultaneously. Couplings were identified between several combinations of parameters, which will be used to design multi-input-multi-output controllers. Diagnostics were also integrated, with a new routine combining estimates of the NTM location from the magnetic equilibrium, ECE and from changes in the NTM amplitude. A density observer combines measurements from interferometery, bremsstrahlung and neutral pressure with a model incorporating information from gas valves, NBI and pellets to detect fringe jumps and give a more reliable estimate of the density profile. The principle is similar to RAPTOR, which has continued to evolve with the inclusion of varying geometry and a model for sawteeth.
In view of the challenges expected for diagnostics at DEMO, alternative methods of controlling the plasma position were trialled. Both reflectometry and ECE demonstrated their feasibility, and identified challenges with the HFS high density front and with cut-off, respectively. Further contributions to control of future machines are covered in the sections on ITED and DEMO. Likewise, new developments on NTM control and the pellet launcher are described in dedicated sections.

8 Core Plasma Physics

8.1 Turbulent Phase Velocities across the LOC-SOC transition

A hypothesis that has been considered for a long time is that the transition from the linear ohmic confinement (LOC) to the saturated ohmic confinement (SOC) regime is due to a change in the dominant turbulence regime from trapped electron modes (TEM) in LOC to ion temperature gradient (ITG) modes in SOC. To date however, experimental investigations have not been able to either confirm or refute this picture. One way to gain additional information on the type of turbulence is to compare the perpendicular velocity (u_{perp}) with the E×B velocity ($u_{E\times B}$); the difference gives the turbulent phase velocity (v_{ph}) that is directed in the electron diamagnetic direction for TEM and in the ion diamagnetic direction for ITG. The recent upgrade of the core charge exchange recombination spectroscopy, which now covers the low and high field side of the tokamak, allows measurements of the poloidal rotation (u_{pol}) in the plasma core and, therefore, measures $u_{E\times B}$ across the LOC-SOC transition with errors in the order of 0.5 km/s.
A comparison of $u_{E\times B}$ with u_{perp} measurements shows a very good agreement across the LOC-SOC transition indicating that v_{ph} is small (<0.5 km/s) and a TEM-ITG transition can not be seen outside of error bars. Additionally, we observe that u_{pol} is directed slightly more in the ion diamagnetic direction than predicted from neoclassical theory.

8.2 Comparison of 2nd Harmonic D Acceleration by ICRF & NBI

ICRF heats D directly at 2nd harmonic in NBI heated discharges. The BC501A neutron spectrometer detects significant energetic tails in the Pulse Height Spetrum (PHS) of a ICRF+NBI discharge, compared to a NBI-only phase. This is predicted by the TRANSP+TORIC+RF-kick (TTR) and SINBAD+SSFPQL+TORIC (SST) code packages, after the successful benchmark on the NBI-only phase. The GENESIS code computes the Neutron Emission Spectra (NES), folded with the response matrix into PHS. TTR retains fast ion orbits, which appear to be the main ingredient for the excellent accuracy, for both neutron rate and PHS. SST predicts three times more neutrons and stronger energetic tails.
NES are unfolded from experimental PHS with two methods, one based on maximum entropy and one on adaptive kernels. The result is similar for both methods, with artifacts due to a feature in the response function at low light output. This can be reduced by increasing the smoothing level, at cost of losing physics information such as the characteristic double-peak shape of the NES.

8.3 Ion Cyclotron Emission

Ion Cyclotron Emission (ICE) is a resonant interaction between the gyromotion of a population of energetic particles and plasma waves. It is mainly observed when fusion products,

neutral beam ions or ICRF-accelerated particles reach the plasma edge and excite Compressional Alfvén Eigenmodes. The emission is passively acquired by radio-frequency probes located either on the vessel walls or inside the ICRF transmission lines and display a feature-rich time-evolving spectrum with peaks at multiples harmonics of the fast ion cyclotron frequency at the edge. From the analysis of these data, it should be possible to determine some characteristics of the barely trapped fast ions (concentration, constant of motions) which are an important vector of energy transport from the center to the edge. This capability and its non-intrusive character make it an interesting candidate for ITER. Thus, the ITPA on energetic particles decided to start a Joint EXperiment (JEX) with AUG, JET, DIII-D, MAST, LHD and KSTAR to investigate the emission under a wide range of plasma and heating conditions. The purpose is to validate the existing theoretical models, to setup a method to reconstruct the fast ions properties from the signal and to design a detector for ITER.
AUG has intensively investigated ICE in the last years and led the setup of this ICE-JEX. The existing diagnostic has been upgraded and is now based on an array of six pairs of $\dot{B}$ probes (for parallel and perpendicular RF component with respect to the magnetic lines). They are presently connected to linear detector and a 200 kS/s digitizer to measure the time evolution of the emission.
Two main results were obtained this campaign: 1) a statistical study on 80 discharges showed that the intensity of the emission is not linearly dependent on the neuron rate (a measure of the fusion product concentration) but presents a saturation. 2) The correlation of the emission with the ELMs shows that the ICE signal can be either in phase or phase opposition with the crashes depending on the plasma conditions. These results will be compared with the data from the other machines and the theoretical models. A FPGA-based digitizer has also been developed to be able to process in real time the spectrum evolution and evaluate the mode numbers of the observed instabilities.

8.4 Measured and Simulated Fast-ion Profiles

The Fast-Ion D_α (FIDA) technique has become a reliable tool to probe the fast-ion velocity space distribution at different radial positions inside the plasma. The FIDA spectrometer has been optimised and the applied tomography algorithms have been improved and successfully tested. The NBI injection peaks can now reliably be resolved with the correct injection energy, which has been tested by comparing 60 keV and 93 keV NBI sources. This made the investigation of 2nd harmonic ion cyclotron resonance heating possible where accelerated beam ions and accelerated thermal deuterium ions were found at different pitch angles. In addition, more detailed studies of the sawtooth induced fast-ion redistribution have become possible. Good agreement with the predicted energy and pitch angle dependence has been found inside and outside the sawtooth inversion radius.
Significant progress has also been made with the interpretation of off-axis NBI experiments. In particular the analysis of beam footprints on the inner wall by infrared measurement, but also studies of the beam emission, exhibited discrepancies between the assumed and actual NBI geometry in the range few centimeters (< 0.2 degrees). This explains the main discrepancies between measured and simulated fast-ion profiles observed previously. Moreover, better agreement is obtained between predicted and measured NBI heat and momentum deposition profiles based on modulation experiments.

8.5 Electrostatic Asymmetries in the W-density

The importance of understanding the behaviour of tungsten in the plasma has motivated a study of the poloidal asymmetries in the impurity density. One of the drivers are the fast minority ions produced by ICRF heating. These ions have a significantly higher perpendicular than parallel momentum and therefore increased magnetic trapping on the outboard side of the plasma. The consequent perturbation in the electrostatic potential affects significantly the poloidal distribution of the highly charged W ions. The new SXR tomography has allowed for observing such asymmetries (figure 20a). The related experiments also provide a stringent test for ICRF heating models. The comparison between TORIC-SSFPQL and TORIC-FFPMOD presented in figure 20b indicate a good match with the experimental asymmetry profile.

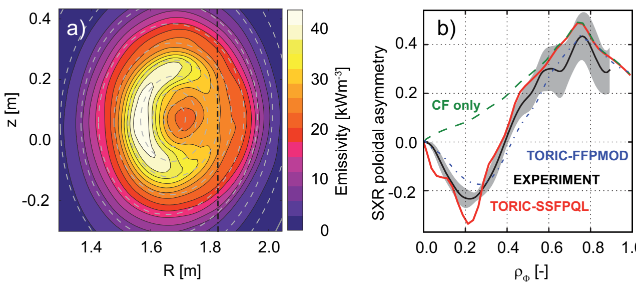

Figure 20: a) SXR profile (#30812, 4.6 s) showing a HFS asymmetry. b) comparison between the measured asymmetry, calculations from the TORIC models and pure centrifugal force (CF).

8.6 Helium as Minority

The presence of helium is fundamentally connected to the performance and success of a fusion reactor. To assess the amount of helium 'ash' in the plasma core of a future fusion reactor, a prediction of the helium density profile shape is required. Hence, an experimentally validated theoretical description of the low-Z impurity transport is necessary. However, at AUG, comparisons of experimental helium and boron density profile gradients with local quasi-linear gyrokinetic simulations revealed discrepancies. While qualitative agreement can be observed, in most cases the theory under- or overpredicts

the experimental impurity density gradients of both impurities. To gain more insight into the low-Z impurity transport properties, further investigations were performed on both the theoretical and the experimental side.
Furthermore, helium has been observed to cause deterioration of the plasma confinement, even at low concentrations similar to those expected in ITER, an effect which is not yet understood. To this end, helium was added in high confinement deuterium plasmas, leading to helium concentrations of up to 12%. A clear reduction of the plasma stored energy of about 10% was observed in both nitrogen-seeded and non-seeded discharges. The analysis is ongoing in order to pinpoint the physics reason behind the observed confinement degradation.

8.7 Comparison of AUG and TCV Intrinsic Rotation Profiles

AUG and TCV intrinsic rotation profiles have been compared within the framework of the European intrinsic rotation database. Interestingly, the toroidal velocity profiles from the two machines are very similar. However, the rotation frequency profiles, which account for machine size, show differences. The TCV profiles display a wider range in ω_ϕ and, more importantly, a wider range in normalised toroidal rotation gradient, u'. A comparison of the parameter dependences of the observed u' values yields encouragingly similar results (see figure 21). The trend with R/L_{ne} is particularly striking. Many core-localised residual stress terms are expected to depend on ρ^*, but preliminary investigation, using the inter-machine comparison, showed no clear evidence of a ρ^* dependence. Ideally, the profiles from the two devices should be compared at matched dimensionless parameters. Obtaining the data for such a comparison, will be the focus of future experimental effort.

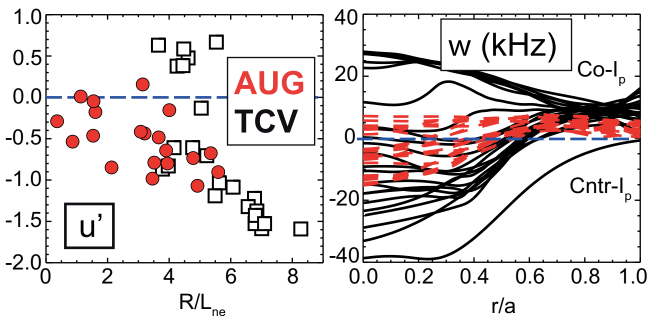

Figure 21: Comparison of rotation gradients and frequencies of AUG and TCV.

8.8 Low-Z Impurity Transport Studies Using CXRS

The plasma performance is highly affected by the impurity concentration and, therefore, a fundamental understanding of impurity transport in fusion plasmas is of great importance. With the CXRS diagnostics at AUG the temperature, density, and rotation velocity profiles of low-Z impurity species can be routinely measured with high spatial and temporal resolution and, thus, provides the necessary tools to perform a detailed investigations of the low-Z impurity transport. Previous work using CXRS has focused primarily on steady-state profiles, which deliver the ratio of the diffusive and convective transport coefficients. However, from the time response of the density profiles after applying an external perturbation the convective and diffusive components of the transport can be separately determined. A new method of achieving this by a less conventional approach that is better suited to the limited time resolution of the CXRS diagnostics is a sinusoidal modulation of the boron density invoked by modulating the power from the ICRF antennas. A feasibility study performed demonstrates the viability of the technique. The transport coefficients are extracted by simulating the experimental phase and amplitude profiles with the transport code STRAHL.

8.9 Influence of H-isotopes on Core Transport and Edge Transport Barrier

A scan of the dimensionless gyroradius ρ^* was performed via a hydrogen isotope scan in L-mode. Detailed measurements, transport analysis and transport modelling are done to investigate the isotope effect on transport which is the break down of the gyroBohm transport scaling when varying the isotope mass.
Despite a near perfect profile match in density, temperature and rotation the confinement time is reduced in hydrogen compared to deuterium consistent with previous studies. The individual analysis of electron and ion transport channels reveals that the electrons are unaffected by the different isotope. The electron temperature is matched with the same electron heat fluxes q_e. In the case of pure electron heating a match of q_e requires higher total heating power in hydrogen because the energy exchange between electrons and ions is more efficient than in deuterium. This leads to a larger ion heat flux in hydrogen. Nonlinear simulations with the GENE code suggest these L-mode discharges are ITG dominated. Additionally, a scan of L_{Te} and L_{Ti} at two different radial positions suggests that a larger ion heat flux itself causes reduced confinement. This is confirmed by experiments which show that the ion energy confinement is lower than the electron energy confinement time. Consequently, the isotope effect of confinement is reduced when matching the electron-ion heat flux ratio more closely.
At the transition of L- to H-mode the physics at the plasma edge becomes more dominant and another consequence of the isotope mass is reflected in the L→H threshold power which is well-known to be about two times higher in hydrogen compared to deuterium. This can be interpreted under the assumption that the radial electric field gradient at the plasma edge drives a sheared perpendicular velocity which stabilizes the turbulence and therefore reduces the transport. The neoclassical radial electric field at the plasma edge being mainly driven by the ion pressure gradient, one may assume that it depends on the ion heat flux and on the heat transport.

Dedicated experiments have indeed indicated that at the transition to H-mode the edge ion pressure gradient profiles in hydrogen and deuterium are very similar whereby the required ion heat flux is about two time higher in hydrogen compared to deuterium. The different observations in L-mode, at the L→H transition and also in H-mode will help towards a better understanding of the influence the isotope mass has on transport properties and the edge stability.

8.10 Plasma Response on MP Using the 3D Boundary Distortion and MHD Models

Best ELM mitigation and ELM suppression at DIII-D as well as at AUG are achieved by MP fields, when the applied poloidal mode spectrum is aligned with the mode (kink) at the edge that is most strongly amplified by the plasma. This kink mode causes a 3D distortion of the plasma boundary, which is static to the applied MP field.

This 3D displacement is measured using data from toroidally localised high resolution diagnostics e.g. ECE and rigid rotating n=2 MP fields with different applied poloidal mode spectra. To interpret even oblique ECE measurements accurately, forward modelling of the radiation transport has been combined with ray tracing. The measurements are compared to synthetic ECE data generated from a 3D ideal MHD equilibrium calculated by VMEC. The measured penetration of the displacement agrees with VMEC, whereas the measured amplitudes on the outer mid-plane are slightly larger. Although the calculated magnetic structure of this edge kink peaks at poloidal mode numbers larger than the resonant components |m|>|nq|, the displacement derived from the ECE-imaging (ECE-I) is mostly resonant. This is expected in ideal MHD in the vicinity of rational surfaces and reproduced by VMEC and MARS-F.

8.11 Approaching the Ideal Kink-limit

The improved H-mode scenario (or high-β hybrid operation) is one of the main candidates for high-fusion performance tokamak operation, which offers potential steady-state scenario. In this case, the normalised pressure β_N must be maximised and pressure driven instabilities limit the plasma performance. These instabilities could have either resistive (2,1) and (3,2) NTM, or ideal character (n=1 ideal kink modes). In AUG, the first limit for maximum achievable β_N is set by NTM. Application of pre-emptive electron cyclotron current drive at the q=2 and q=1.5 resonant surfaces reduces this problem, such that higher values on β_N can be reached. Experiments have shown that, in spite of the fact that hybrids are mainly limited by NTM, proximity to the no-wall limit leads to amplification of external fields that strongly influences the plasma profiles: for example, rotation braking is observed throughout the plasma and peaks in the core. In this situation, even small external fields are amplified and their effect becomes visible. Analysis of the plasma reaction to external perturbations allowed us to identify optimal correction currents for compensating the intrinsic error field in the device. Such correction, together with analysis of kinetic effects, will help to increase β_N further in future experiments.

8.12 Growth of the (1,1) Mode before a Sawtooth Crash

Sawtooth oscillations are a periodic relaxation process of the plasma temperature, density and other plasma parameters in the central region of a tokamak. This relaxation process is associated with growth of internal (1,1) kink mode, which leads to the crash event in the plasma core. A new method was developed to derive the core displacement, growth rate and frequency of this (1,1) mode before the sawtooth crash. The method uses soft X-ray tomography to trace the displacement of the plasma core due to this mode with high accuracy. It was shown based on these measurements that the sawtooth precursor has strongly non-monotonic growth. For typical cases, it starts with a smaller growth rates and reaches its maximum just before the crash. At the same time, there is a wide variety of different sawtooth crashes which show different behaviours. The crash time itself is also varies from tens to hundreds of microseconds in these. Presented method provides convenient measurement of the (1,1) mode amplitude which can be used for a direct comparison between experiment and non-linear MHD simulations of the sawtooth crashes.

8.13 Seeding of Tearing Modes via MP in Low ν* L-modes

AUG is equipped with a set of 16 in-vessel saddle coils. These coils enable the generation of resonant and non-resonant, static and rotating MP fields. By applying a slowly rotating (0.5 Hz) n=1 MP field ramp three phases can be distinguished:

The phase of linear plasma response, where the MP field induces a deformation of the plasma but is not large enough to drive magnetic reconnection. At the beginning of the second phase the MP field exceeds a threshold which initiates forced reconnection. The related formation of a magnetic island is detected in the magnetic measurements and in the electron temperature profile. The plasma rotation was measured within beam blips. After the mode penetration a drop of the core rotation was observed in all discharges. In ECRH assisted discharges also the edge rotation decreased whereas it spins up in the purely ohmically heated discharges. After the mode onset the island grows non-linearly to a size of several centimetres until minor disruptions occur which mark the beginning of the third phase. In this third phase the island is entrained by the MP field. However it s observed that the island already starts to rotate slightly slower (<0.5 Hz) than the MP field due to additional torques acting on an island with finite size. In addition in agreement with theory also the calculated electron flow at the resonant surface (electron perpendicular velocity) is as small as the island rotation.

8.14 Advanced MHD Real-time Control Processes

For ECCD-based NTM control, a new technique pioneered at TCV has been implemented. The optimal ECCD deposition location is only available with some finite uncertainty. With continuous sweeps through this position, one gets periodical 'hits' of the optimum and can utilise the fast response of the NTM. In an experiment (figure 22), stabilisation occurs immediately when the island position is reached by three combined ECCD beams (t=5.43 s, 2.2 MW). This result agrees with the prediction from theory that the NTM will be stabilised when ECCD driven current exceeds the bootstrap current by more than 20%.

In experiments on integrated control, linking β-control to the detection of NTM proved to be a reliable recipe for high performance discharges. Without an NTM, β is continuously ramped up. When an NTM is detected, the NBI power is kept constant and an actuator management routine allocates free gyrotrons to stabilise the NTM. In one case even a single gyrotron was able to shrink the NTM amplitude significantly while β_N is restored to above three for several confinement times. The efficient NTM stabilisation indicates good accuracy from the new combination of three independent estimates of the NTM location: from equilibrium, ECE and changes in the NTM amplitude.

The avoidance of the L-mode n_e-limit in 2014 has been extended to the H-mode. Typically the MHD occurs at such high densities, that the MHD triggered ECCD fails due to cut-off. Therefore a pre-programmed time for the ECCD has been used based on the characterization of H→L transition in an operational diagram. The disruption could be avoided and the discharge returns into H-mode with good confinement. In high β_N scenarios the feedback controlled application of rotating MP fields could entrain a locked (2,1)-mode. For MHD detection the new SVD based algorithm operates routinely both on- and off-line.

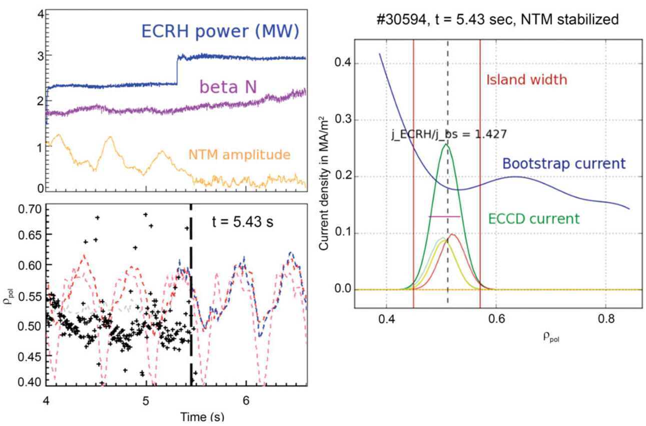

Figure 22: Sweeping motion of three gyrotrons across an NTM location (black crosses).

8.15 High Radiation Scenarios

Power exhaust is one of the major challenges of future tokamaks. The dissipated power fraction has to be in the order of 90-95%. A possible solution is highly radiative scenarios, where impurities are deliberately injected into the plasma. By the choice of the seed impurity, the location of the radiation may be controlled, i.e. divertor radiation with N or Ne seeding, SOL radiation with Ar seeding or radiation in the pedestal region with Kr.

In recent experiments, the different seed impurities were tested at ITER relevant power fluxes ($P_{sep}/R \leq 12$ MW/m), using the maximum available heating power of up to 26 MW. Ne and Kr proved to be not applicable as seed impurities at the exhaust-relevant parameters for AUG due to their detrimental effects on the plasma pedestal. With intense seeding of N or Ar, fully detached divertor plasmas, where only marginal heat fluxes reach the divertor target, are achieved at the highest heat fluxes. The radiated power fraction is up to 90%. For both impurities, it is observed that the dominant radiation originates from inside the confined region, while the stability of the discharge is not affected and the energy confinement is only slightly reduced. The location of the radiating region relative to the X-point location can be actively controlled via changes in the heating power or impurity seeding level.

8.16 Intermittent Turbulent Fluctuations in I-mode

The I-mode confinement regime is characterised by steep edge profile gradients in the electron and ion temperatures, but not in the density. Probably connected to this peculiar behaviour is a quasi-coherent feature in the fluctuation characteristics called the weakly coherent mode (WCM), which resides in the plasma edge. Recently, strong fluctuation amplitude bursts have been found in the I-mode edge plasma. The structures appear only in I-mode, and they become stronger with increasing confinement quality. They have higher amplitudes than the L-mode fluctuations (figure 23) and are strongly intermittent. It was demonstrated that the density fluctuation bursts are linked to the WCM. This is a hint that they might play a role in inhibiting the density profile to steepen in I-mode.

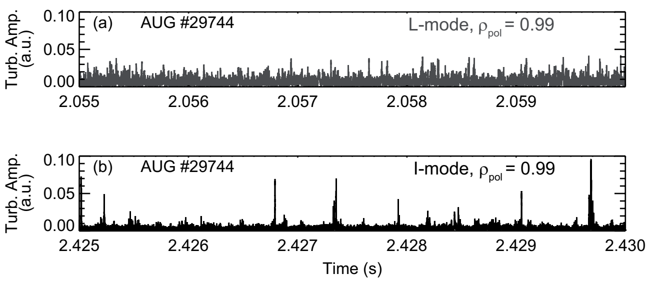

Figure 23: Turbulence amplitude time traces for L- (a) and I-mode (b).

8.17 Turbulent T_e Fluctuation CECE Measurements Compared to Simulations

A fusion reactor's size is dominated by its turbulent heat transport. Understanding and predicting the turbulence is thus of high priority. In collaboration with MIT, a Correlation Electron Cyclotron Emission (CECE) radiometer was installed for measuring turbulent δT/T up to $k_\perp$~0.76 cm^{-1}. CECE has a much greater sensitivity to small scale fluctuations than a conventional radiometer. δT/T profiles have successfully measured for the first time on AUG in both helium and deuterium. Figure 24 shows a comparison of GENE gyrokinetic simulations with our experimental results.

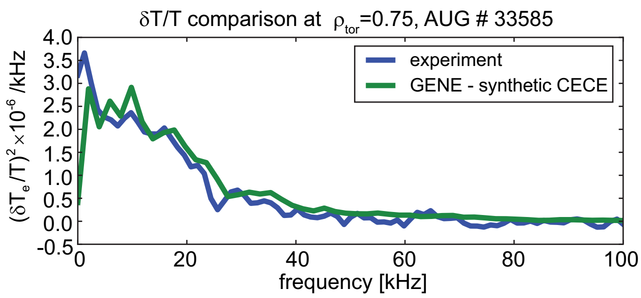

Figure 24: GENE and experimental results show good agreement.

The small wavelength limit will be improved to 1.4 cm^{-1} in 2017 with a new focusing mirror, doubling our sensitivity. The number of channels will also be increased from 10 to 20, allowing correlation lengths and more detailed profiles to be measured.

8.18 New n_eT_e Cross-phase Prototype Diagnostic

Plasma turbulence is an intrinsically multi-field phenomenon. Not only the fluctuations themselves affect particle and heat transport, but also the relationships between them. In order to study the phase relationship between T_e and n_e fluctuations prototype diagnostic which couples a reflectometer and Correlation ECE radiometer has been constructed.

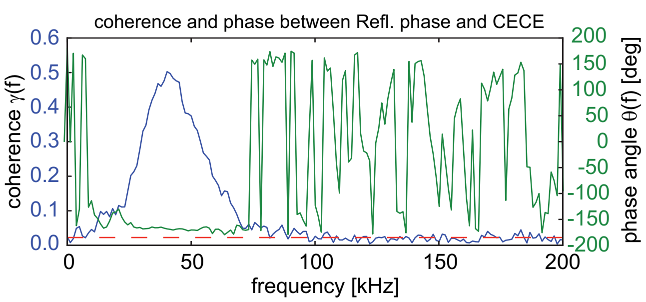

Figure 25: Correlation and cross-phase between reflectometer phase and CECE radiometer.

The radial alignment between the radiometer cut-off and the ECE harmonic must be within the turbulent correlation length, which can be very challenging. Nonetheless, successful trials of the diagnostic in the plasma edge region have been carried out, shown in figure 25.

A new focusing mirror and a doubling of CECE channel number in 2017 will make the plasma core more accessible. A synthetic diagnostic for quantitative comparison with GENE simulations will be developed.

8.19 Disruption Mitigation

The range of neon quantities injected in AUG plasmas during MGI experiments has been extended below the minimum quantity necessary for disruption mitigation and the influence of these quantities on the induced plasma-shutdown have been analysed and documented. 1-D modelling of the pre-TQ phase, performed with the code ASTRA-STRAHL, reproduces its duration and the parametric dependences observed experimentally. The observed normalized CQ duration is in the prescribed ITER range over a wide range of injected impurity amounts. Forces and heat load are mitigated in AUG with >10^{21} atoms of neon. At lower quantities of gas the forces increase towards the unmitigated values. Thermography data provide experimental indication that more energy is deposited onto the divertor plates during the TQ as the gas amount is decreased. The problem of characterizing the thermal loads during disruptions has been and remains complex because of the 3-D nature of the energy flow and radiative dissipation. Experiments on runaway electron formation and losses have been performed. The quantity of argon injected, to create and then suppress the RE, has a clear influence on the time decay of the RE beam, i.e. the more impurities, the faster the RE current decay. The critical electric field, calculated from the electron density and the concentration of argon, is larger than the toroidal electric field sustaining the RE beam, and thus significantly accounts for the RE energy losses associated with the observed current decay. Large error bars affect the estimated argon concentration because of uncertainties in the radiation coefficients.

8.20 Li-pellet Injection

Several tokamaks reported performance improvement with the presence of lithium (Li) in the plasma. At AUG experiments were performed to find out if such effects can be observed also when operating with an all-metal-wall. To keep the Li amount low and minimize the impact on the first wall, Li pellets providing a penetration deep into the plasma were used. A gas gun launcher was developed capable of injecting pellets containing $1.6 \cdot 10^{20}$ Li atoms at up to 2 Hz. The pellet speed of 600 m/s turned out to be sufficient to achieve core penetration and to create an almost homogeneous Li concentration in the plasma of up to about 15%. The Li sustainment time in the plasma decreased with increasing heating power from 150 to 40 ms. Hence, only transient Li presence without any pile up in the plasma was achieved. The Li impact on the confinement was investigated mainly in a plasma scenario with proven positive sensitivity to nitrogen seeding. Injection of a Li pellet however resulted always in a small transient reduction of the plasma energy. In pure D plasmas this loss decreased with increasing plasma energy until it almost vanished for the highest applied heating power of about 15 MW. We attribute this behaviour to higher Li induced radiation losses in lower energy plasmas, essentially caused by a longer Li confinement time. Taking into account radiation losses and the energy consumption required for Li ablation and ionization, the dedicated high power D discharges showed indication for small (2%) increase of the confinement. In N seeded discharges, a significant loss of the N induced confinement surplus always took place after pellet arrival.

8.21 Collective Thomson Scattering (DTU Denmark)

The DTU activities at AUG in 2015 and 2016 included both diagnostic development of the collective Thomson scattering (CTS), the Li beam, and probe diagnostics, together with a number of physics studies in the field of fast-ion transport, radiation transport, SOL physics and turbulent transport.

The CTS system, operated by the DTU group, was upgraded in 2015/16 to operate in the vicinity of 140 GHz. This allowed scattering off the X2-heating ECRH beam in piggy back for CTS development. Here successful measurement of the bulk ion temperature and drift velocity were obtained and a systematic comparison between CTS and CXRS was initiated. The hardware upgrade also paved the way for a joint IPP/DTU collaboration on detection of ECRH accelerated non-thermal electrons together with a joint PhD project on the study of anomalous parametric decay of X2 ECRH. Experiments mapping out the fast-ion re-distribution due to sawtooth crashes measured simultaneously by FIDA and CTS were continued within the EUROfusion framework. The information from both diagnostic has been combined by velocity space tomography and a 2D fast-ion distribution function resolved in energy and pitch has been reconstructed just before and just after the crash.

DTU also contributes to investigations of turbulence and transport in the SOL. The investigations are partly based on probe measurements at the outboard mid-plane and the divertor in collaboration with RFX ENEA (Padova) and ÖAW (Innsbruck). The measurements are augmented with numerical simulations applying the global SOL turbulence code HESEL. This adds an extra dimension to the understanding of the underlying SOL-physics. HESEL is embedded in a Kepler workflow facilitating the direct comparison with experiments. Additionally, HESEL is equipped with synthetic diagnostics – e.g., probes, Li-Beam spectroscopy, Gas Puff Imagining, facilitating the interpretation of the experimental observations including interpretation and operation of the diagnostics and improving the understanding.

8.22 Reflectometry (IST Portugal)

Physics studies: (i) High density front formation: Density profile studies revealed that the region of high density formed in the inner divertor expands to the midplane, leading to strong poloidal asymmetries in the SOL density. The evolution of the midplane density profile in the HFS was observed to respond to divertor oscillations, changing significantly for different detachment states. The HFS/LFS density asymmetry was found to increase with input power and to decrease with N_2 seeding. (ii) Density fluctuations: $\delta n/n$ radial profiles were obtained for L-mode showing a HFS/LFS asymmetric response to different magnetic configurations (fluctuations increase at HFS and are reduced at LFS as the configuration is changed from LSN to USN). In addition, the turbulence behaviour displays pronounced variations correlated with the evolution of divertor detachment.

Diagnostic developments: (i) New multichannel reflectometer system: A new multichannel X-mode reflectometer system has been successfully installed and commissioned to measure the edge electron density profile in front of the ICRF antenna. The plasma can be probed using ten pairs of antennas installed at different poloidal locations inside the antenna. The analysis software necessary to reconstruct the density profiles was also developed and data validated against other diagnostics. (ii) Plasma position control using reflectometry: The viability of using microwave reflectometry for controlling the position of fusion plasmas has been demonstrated for the second time. In this new experiment, a reflectometer at the HFS was used in addition to the one at LFS originally used. The development of this technique is essential for long pulse devices, where the standard magnetic based control might prove unreliable.

8.23 New Real-time Diamagnetic Flux Diagnostic

Real-time diamagnetic flux measurements are now available. To achieve the highest possible sensitivity, the diamagnetic measurement and compensation coil integrators are triggered shortly before plasma initiation when the toroidal field coil current is close to its maximum. The integration time can then be chosen to measure only the small changes in flux due to the presence of plasma. Two identical plasma discharges with positive and negative magnetic field have shown that the alignment error with respect to the plasma current is negligible. The measured diamagnetic flux can then be compared to that predicted by transport code simulations (TRANSP) and from magnetic equilibrium reconstruction codes. The diamagnetic flux measurement and prediction can be used together to estimate the coupled power in discharges with dominant ion cyclotron resonance heating. The prediction of diamagnetic flux is overestimated in discharges with MHD modes that lead to fast ion pressure losses.

9 Edge and Divertor Physics

9.1 n_e Profile and Turbulence Evolution during L→H Transitions Studied with the Ultra-fast Swept Reflectometer

During the 2015-2016 campaign, density turbulence behaviour has been investigated using a dual-band (V & W-band: 50-104 GHz) Ultra-Fast-Swept reflectometer on loan from CEA. For the first time, the 1 µs sweep time (plus dead time of 0.25 µs) allows electron density profiles, as well as turbulence levels and the turbulence radial wavenumber spectrum to be resolved over fast plasma events, such as the L→H transition and I-phases – where a significant reduction in the turbulence level at small k_r are observed. Combining UFSR and ECE measurements, the neoclassical radial electric field E_r has been estimated. During the I-phase its value oscillates with the frequency of 1-2 kHz and continuously deepens around the E_r-well reaching values of -30 to -40 kV/m. The oscillations in E_r and the density fluctuation level in the pedestal region have been correlated during the high-density I-phase. The absence of a relative phase shift supports an I-phase description based on ELM-like crashes: with growing pressure the enhanced background E_r shear suppresses the turbulence, while after a crash the weak background E_r leads to the turbulence increase.

9.2 Characterisation of inter-ELM Magnetic Oscillations

In highly confined tokamak plasmas periodically appearing ELMs are accompanied by mode-like MHD activities with defined toroidal mode numbers. These ELM associated fluctuations might play an important role for the onset of the ELM crash, as their structure seems to be conserved during the crash.

On AUG these fluctuations were characterised in terms of their mode numbers and their location in the pedestal by magnetic pick-up coils. It was found that typical toroidal mode numbers of these modes are in the range of n=-1…-12, where the minus sign means that they rotate in the electron diamagnetic direction in the laboratory frame. Several modes appear at the same time with different rotation velocities at different positions in the strong gradient region at the edge with a separation <2 cm and the development of these modes in terms of their mode numbers is correlated with the development of the pedestal parameters.

9.3 Stochastic Layer Properties Induced by MP via Heat Pulses

MP fields are studied in AUG and other tokamaks due to their effects on ELMs. MP leads to a stochastic layer near the edge with open field lines. However, the plasma response field can shield the MP, leading to a narrowing of this layer. A new experimental method to estimate the stochastic layer width was applied in an AUG L-mode discharge with MP. The method relies on depositing ECRH pulses in the plasma edge while measuring the divertor target heat flux with IR thermography and Langmuir probes. The results were compared to simulations of the time dependent heat pulse propagation on a constant plasma background with the EMC3-Eirene code package, using an ad-hoc screening model. In the simulation with MPs, but without screening, the heat pulse propagation time was considerably shorter than in the simulations without MP, which is due to the parallel transport along the open field lines. Comparing the MP and the non-MP discharges experimentally, approximately the same propagation time was found, indicating strong screening.

9.4 2D Heat Flux L-mode with MP Fields

The influence of external MP fields onto the heat flux transport in the SOL was investigated in L-mode discharges. A slow rotation of the external perturbation field of 1 Hz allows to measure the complete 2D heat flux structure at the divertor with constant background plasma conditions using infrared thermography which observes a fixed toroidal position. Applying MP with a phasing aligned to field lines at the edge (so called resonant configuration) results in a pronounced change of the heat flux pattern compared to the axisymmetric case without MP for low plasma density. As shown in the figure 26 a lobe pattern is present, with a toroidal periodicity of the applied n=2 perturbation. Nearly no impact on the heat flux pattern is observed using a phasing opposite to the resonant configuration (non-resonant configuration). The constant L-mode conditions allow to average the heat flux in time, corresponding to an average over the whole toroidal circumference of the divertor. Fitting the diffusive 1D model (Eich-Fit-Function) to the axisymmetric and the toroidal averaged profiles reveals that the defining transport qualifiers, the power fall-off length λ_q and the divertor broadening S, do not change with the application of the MP. The helical structure, however, leads to a toroidal asymmetric heat distribution with enhanced toroidal asymmetry of the heat flux in regions further away from the strike line.

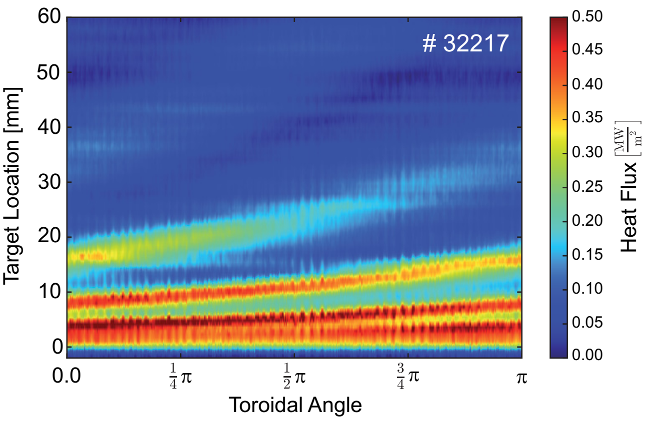

Figure 26: 2D divertor heat flux profie for #32217 with rotating MP (n=2, 1 Hz) measured with a new IR-system.

9.5 Density Dependence of SOL Power Width in L-mode

The density dependence of the scrape-off layer (SOL) power width has been studied in L-mode. The heat flux profile on the divertor target is described by two main quantities, the upstream power fall-off length λ_q and the divertor power spreading S. Figure 27 shows normalised heat flux profiles on the outer divertor target for different plasma densities. It is observed that for the outer divertor target the power fall-off length λ_q has no significant density dependence.

The profile broadening with increasing density is due to the divertor broadening S in the divertor region. It is found that the finite ion gyro radius needs to be included in the evaluation. Doing so results in a temperature and density dependence of the diffusive term of S which is close to the values expected for parallel electron conduction and a temperature and density independent perpendicular heat diffusivity. For the inner divertor a strong dependence of the power fall-off length λ_q on the density is observed. For low densities the λ_q on the inner divertor target is smaller compared to the outer divertor target. With increasing density λ_q on the inner target increases strongly reaching values larger than observed on the outer divertor target. The increase of λ_q occurs while the total power arriving on the inner divertor target stays constant, volume dissipation can be neglected.

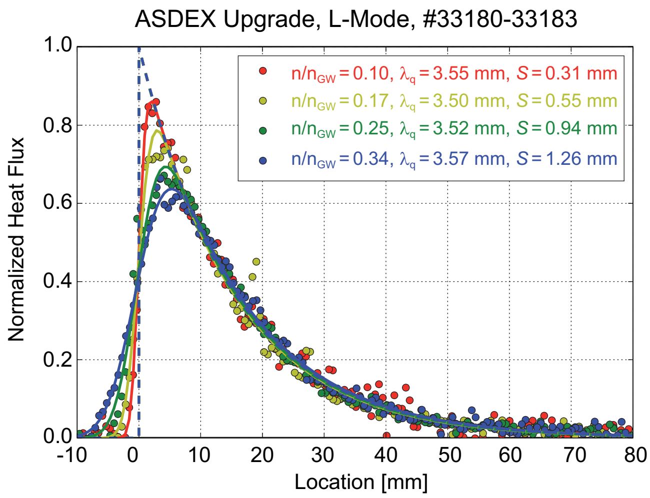

Figure 27: Normalised target heat flux profiles on the outer divertor target for different plasma densities.

9.6 Scaling of Divertor ELM Energy Fluence during ELM Mitigation

The mitigation of ELM induced divertor load is the main objective for the application of external MP in ITER. Extensive studies have been performed to scale and predict the ELM induced divertor load towards large fusion devices. A recent succesful attempt uses the pedestal pressure as the main quantity to scale the parallel ELM deposited energy density, also known as ELM energy fluence, $\varepsilon_{||}$. A model has been derived which is able to describe the observed minimum $\varepsilon_{||}$ in AUG, JET and MAST.

$$\varepsilon_{||} = \Delta_{EQUI} \cdot 2\pi a_{geo} \cdot \sqrt{\frac{1+k^2}{2}} \cdot \frac{3}{2}\, n_e^{ped}\, k_B\, T_n^{ped} \cdot \frac{B_{tor}}{B_{pol}}$$

Where Δ_{EQUI} is a geometry factor, which is ~1.9 for AUG, a is the minor radius, κ the elongation, $n_{e,ped}$ and $T_{e,ped}$ the electron density and temperature at the pedestal top. B_{tor} and B_{pol} are the toroidal and polidal magnetic field.

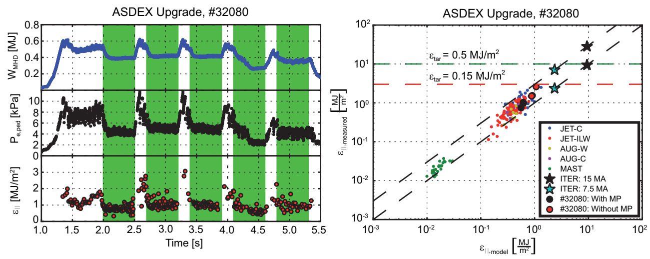

Figure 28: Influence of MP on pedestal pressure and ELM energy fluence.

Figure 28 (left) shows the time trace of the stored energy W_{MHD}, pedestal electron pressure $p_{e,ped}$ and the ELM induced parallel energy fluence $\varepsilon_{||}$ of an ELM mitigated discharge (#32080, B_{tor}=-2.5 T, I_p=0.8 MA, P_{aux}≈7.5 MW). The green shaded areas indicate the periods in the discharge with active external magnetic perturbation. It is seen that all three quantities are reduced during phases with external magnetic perturbation compared to unperturbed phases. The reduction of ELM induced energy fluence $\varepsilon_{||}$ is correlated to the reduction of the pedestal pressure $p_{e,ped}$. Figure 28 (right) shows the measured parallel ELM energy fluence in comparison to the model prediction. A large data set from unmitigated discharges AUG, MAST and JET are shown for comparison. It is seen that both cases with and without external MP are captured within the prediction of the model. The observed reduction of the ELM energy fluence in the presence of external MP is explained largely by a reduction in pedestal pressure. Studies for ELM mitigation should hence correct the assessment of mitigation for the reduction of the pedestal pressure.

9.7 Characterisation of Pulsations Close to the L→H Transition

At the transition from L- to H-mode regimes, characteristic pulsations of plasma flows and turbulence, so called limit-cycle oscillations (LCO), can occur at the edge of a fusion plasma. In new experiments at AUG it was found that the modulation of flows and gradients during LCOs is accompanied by a strong magnetic activity. Figure 29 shows the result of a cross-correlation analysis indicating an up-down asymmetry of the poloidal magnetic field perturbations during the LCOs. The magnetic activity increases during the development of an edge pedestal, and is preceded by type-III ELM-like precursors. These observations together with the frequency scaling of LCOs seem to be inconsistent with prevailing electrostatic LCO models based on a predator-prey interaction between zonal flows and turbulence.

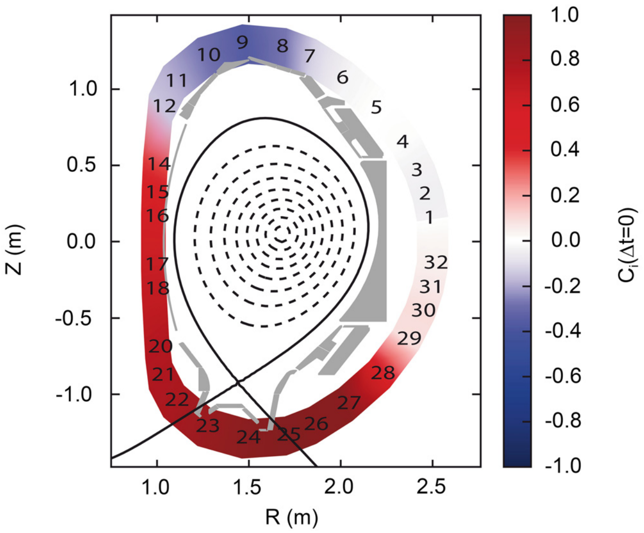

Figure 29: Poloidal mode structure of MP during LCO at the L→H transition.

9.8 Density Shoulder Formation during H-modes

In recent years, the understanding of SOL transport was substantially improved by relating the flattening of the density profiles observed in many tokamaks – known in the literature as 'shoulder formation' – with the enhanced perpendicular transport caused by a change of regime of far SOL filaments taking place when they are electrically disconnected from the wall due to collisionality. In the 2015-2016 campaign, a number of experiments have been carried out in order to check the validity of this mechanism for the reactor-relevant H-mode. Two main results have been achieved: First, the formation of a shoulder has been confirmed in inter-ELM H-mode plasmas. Second, the link between shoulder formation and filament transition has also found to remain generally valid, although the physical picture complicates: collisionality has a similar effect on filaments as in L-mode, and it remains a necessary condition for the shoulder formation in H-mode. However, it is no longer a sufficient one, as a second threshold appears related to the level of deuterium fuelling.

9.9 Stability and Propagation of the HFS Detachment Front

During detachment a structure of strongly enhanced density develops close to the inner target. Its dynamics is approximated by those of radiative fluctuations and has been studied by means of time-delay-estimation. The dynamics can be described as follows: at increasing density the ionization front moves upstream to reduce ionization radiation in order to balance the increased recombination radiation. The recombination zone stays close to the target strike point. The perpendicular neutral motion determines the dynamics. The divertor nose constitutes an obstacle for the perpendicular neutral flux from the target to the region above the X-point. Passing into this shadow the neutral flux above the X-point is strongly reduced, the ionization front fades away and the heat flux from upstream can increase the temperature in the recombination region, subsequently reducing recombination and reforming an ionization front below the X-point. A cyclic reformation of the ionization front propagating from below to above the X-point occurs leading to a fluctuation as observed in the experiment.

9.10 Stability of Solid-W Divertor Tiles

In 2013 a redesigned solid-W divertor, Div-III, was installed. In two experimental campaigns more than 3000 plasma discharges with up to 10 s duration, 110 MJ plasma heating, and reaching a P_{sep}/R of 10 MW/m were conducted. Div-III is an adiabatically loaded divertor with tungsten target tiles clamped onto a water cooled structure. The tungsten target tiles were inspected and characterized during the shutdown period in 2014. As a result of the target inspection characteristic modifications of the plasma exposed tiles were detected that were not found during the target qualification in a high heat load test facility. 126 out of 128 tiles reveal deep cracks through the target. A lot of tiles show shallow cracks in the high heat load region and finally protruding tiles with strong local damages were found. It should be noted that none of these target damages have caused an unscheduled opening. The deep cracks where neither found in GLADIS tests nor in tungsten prototype targets exposed in 2011/2012. The mechanism for crack formation in Div-III is under investigation with FEM-modelling and high heat flux test.

9.11 Arc Erosion of Full-metal PFC at the Inner Baffle Region

In the standard picture of plasma wall interaction, arcs are not taken into account as erosion source: in the 80 s it was shown that for carbon physical and chemical sputtering dominate, and arcs are only triggered during unstable plasma phases with enhanced MHD activity. On the other hand arc traces are observed in all present fusion devices. Due to the transition to metal plasma facing components (PFC) the role of arcing on erosion has to be reconsidered again. Recent investigations show that ELMs may trigger arcs at some locations and the morphology of a major fraction of dust collected can be explained by droplet production in arcs. At the inner baffle of the divertor massive polished inserts of tungsten and P92 steel were installed to measure the erosion by arcing (figure 30). As this region is deposition dominated the deposits were removed by wiping to allow measurements at the insert itself. Whereas in the optical picture typically a third of the surface is affected for both inserts, height profiles show strong differences. For tungsten most of the traces are less than 0.4 μm deep and a similar amount of tungsten is deposited close to the traces. These arcs show only a redistribution but no net erosion. Few craters up to 4 µm resulting in an average erosion rate of $2 \cdot 10^{13}$ at $cm^{-2}s^{-1}$ are observed. The behaviour for P92 steel is quite different: most of the traces are more than 4 µm deep, up to 80 µm were observed. The average erosion rate of $400 \cdot 10^{13}$ at $cm^{-2}s^{-1}$, i.e. more than a factor of hundred higher compared to tungsten, was observed. To get an idea of the relevance of the erosion rates, the erosion by arcing at the inner baffle is compared with the tungsten erosion at the outer strike point, the region with the highest erosion rate. As the baffle region is 7 times larger than the strike point region the tungsten release by arcing is 25% of the strike point erosion. For a full P92 steel baffle region about 45 times more iron will be released than tungsten from the outer strike point. Therefore, erosion by arcing has to be taken into account to determine the optimal material mix for future fusion devices. For Be the melting temperature is close to steel, but the effusivity is even higher than for W, which complicates predictions. Further investigations, using different materials are started to disentangle the different effects and to allow estimating the droplet production by arcing.

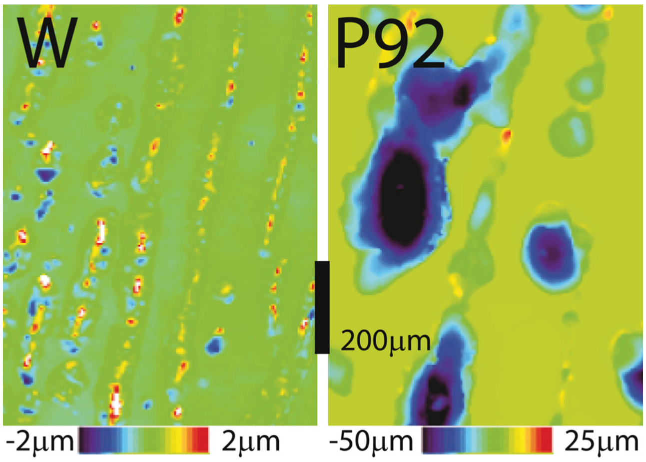

Figure 30: Arc traces in W and P92 steel.

10 Microwave Technology for ECRH (Univ. Stuttgart)

At AUG the new plasma heating system ‘ECRH-3’ is built, which consists of 4 gyrotrons (2 frequencies 105/140 GHz, power 1 MW, pulse length 10 s), individual matching optics, corrugated HE11 waveguide for transmission to the plasma, and steerable reflector antennas. IGVP contributes in the design and construction of components for transmission and related diagnostics.

In 2015, the design of the surfaces of the mirrors in the matching optics box has been finalised, and specifications for the machining of the surfaces where provided. The surfaces of the polarisers were machined, and tested in a 3-mirror resonator. From these tests, rectangular-groove parameters were derived, which can be used as input for polarisation settings. Novel two-frequency directional couplers, which are integrated in the surfaces of mitre-bend mirrors, were developed and specified. These couplers employ leaky wave antennas, basically designed for 105 GHz; in addition, an amplitude grating overlaid on the hole coupling structure produces a receiver lobe for 140 GHz (see figure 31).

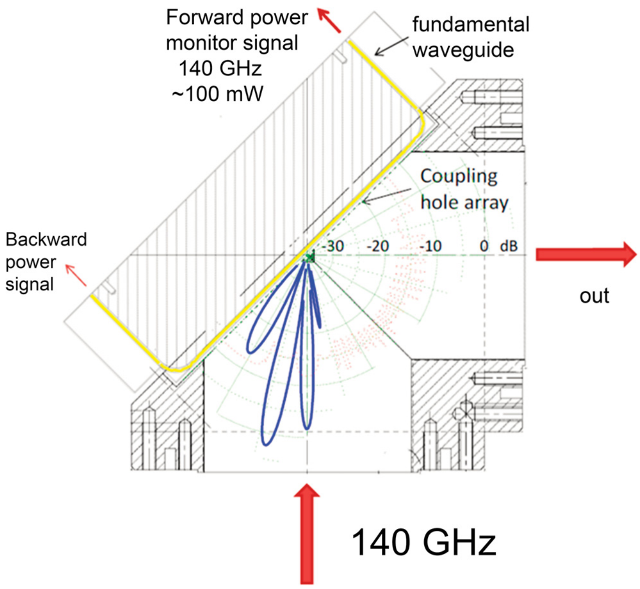

Figure 31: Design of the power monitor integrated in a mitre bend, with the 140 GHz antenna pattern of the coupling hole array. For 105 GHz, the main lobe points into the direction of the incoming HE11 mode.

A 1 MW, 10 s calorimetric load to be used for power measurement of the four gyrotrons via switching mirrors was designed and constructed. The design for compact 1 s absorbers, individual for the gyrotrons, was started. The MC III diplexer, which is optimised for experiments on in-line ECE, is ready for integration into a transmission line; the in-waveguide polarisers are presently equipped with motor drives.

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JET Cooperation

Head: Dr. Thomas Eich

Recent IPP contribution for JET covered various fields. Physics studies on power exhaust by nitrogen seeding were continued. Complementary the ammonia production and retention was measured. A long-term project on data interpretation based on Bayesian methods was finalized. New ground was taken in the field of the effect of Helium on the plasma performance. Modelling of the plasma edge investigated the role of electric fields for LH transitions.

Overview on IPP Involvement

Despite various technical problems with the neutral beam heating system both the Deuterium and Hydrogen campaigns were finalized, though partly with reduced success when compared to previous years. However, the visibility of IPP contributions to JET on international conferences was excellent. In total 22 IPP scientists were seconded to JET in 2016, leading to a total of ~4 ppy of on-site support for the operation of JET. Two IPP scientists were almost permanently on site, being involved in the management of the JET Task Forces. Four long-term secondments of IPP staff were active in 2015/16.

The Effect of Helium on Plasma Performance

The accumulation of fusion-produced helium in the plasma is a potential concern for a future fusion reactor, as it leads to the dilution of the fusion fuel. Furthermore, helium might have a negative effect on plasma confinement. While it is well known that confinement in helium plasmas is lower than in deuterium plasmas, recent experiments have shown that helium degrades confinement already at low concentrations in deuterium plasmas. At JET, helium was deliberately injected in baseline scenario plasmas. The helium concentration was scanned in the range of reactor relevant concentrations, up to a value of 10 % from pulse to pulse. A linear reduction of the plasma energy with increasing helium concentration was observed, as shown in figure 1. Notably, the pedestal pressure remained roughly constant with increasing helium concentration, whereas the core pressure dropped. The ELM frequency was found increasing with increasing helium concentration. Also, a strong reduction in neutron rates was observed, which cannot be fully attributed to the dilution.

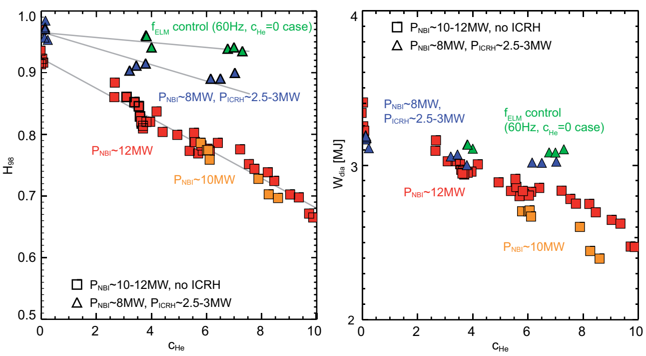

Figure 1: Effect of helium on the plasma confinement: H_{98} factor and diamagnetic stored energy plotted against the helium concentration, in pulses with only NBI heating (squares), with a mixture of NBI and ICRH heating (triangles) and with the employment of ELM frequency control (green triangles).

A reduction of the measured neutrons from the core is explained by the stronger attenuation of the injected neutral beams due to an increasing edge density with increasing helium concentration and, therefore, the produced beam-target neutrons in the plasma core are significantly reduced. The neutral beam attenuation and, thus, the reduced power deposition in the confinement region is sufficient to explain the reduced plasma stored energy. When part of the neutral beam heating power is replaced by wave heating, the reduction in the plasma stored energy is not reduced as drastically as in the NBI-only heated pulses. With the ICRH heating, the temperature profile is kept closer to that of the no helium case. Moreover, when the ELM frequency is controlled to the frequency of the reference pulse without helium by adjusting the deuterium puff, the negative effect of helium on the plasma stored energy almost vanishes.

ICRH Studies at JET

Experiments in JET-ILW showed that the ITER-like antenna (ILA) ICRF antenna has about 20 % lower radiated power from the bulk plasma and about 20 % lower tungsten (W) content compared to the A2 ICRF antennas. This is thought to be due to an improved antenna design that produces a reduced local electric field. The reasons for the increased W levels during ICRF operation at JET are still unclear. Attempts to identify W sources responsible for this, however, were not yet conclusive. At the same time is has been well understood, that the enhanced beryllium sputtering is due to flux tubes connecting PFCs at the high field side to the active antennas. This effect was quantitatively well modeled by erosion codes, confirming that elevated ICRF-specific sheath voltages exist is non-local.

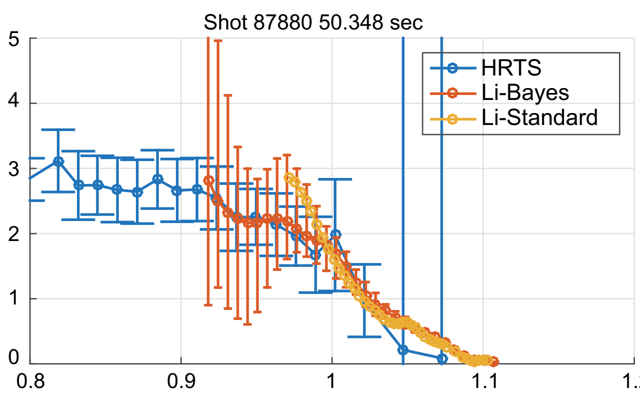

Figure 2: Electron density profile with error bars inferred by the Minerva Li-beam model.

Bayesian Methods for JET

A full spectral Bayesian model of the Li-beam diagnostic for inferring edge density profiles was implemented. The model fully accounts for instrumental effects and the photon statistics, and uses Gaussian processes to model the electron density profile. The spectral background level is dominated by Bremstrahlung radiation and could in the future be used to infer local effective charge at the plasma edge. A Markov Chain Monte Carlo scheme is used to explore the full posterior of the electron density profile (see figure 2) and the calibration factor, the latter is inferred automatically by the model.

Continued Studies on High Radiation Scenarios at JET

The aim of high radiation scenarios is to significantly reduce the power flux into the divertor by high radiation from seeded impurities. This leads to the detached divertor operation, where the plasma is not in direct contact with the divertor targets anymore. Previous experiments with either N, Ne or Ar seeding showed an upper limit of the radiated power fraction of 75 %. Recent experiments applied Kr seeding to further increase this number. In presence of Kr seeding, the plasma undergoes a cyclic transition between an H-mode phase with a detached divertor and an L-mode phase with an attached divertor. The cyclic behavior can be explained by the radial transport of Kr, which is pointing inwards in the H-mode phase and outwards in the L-mode and hence leading to transitions between H- and L-mode.

Ammonia Formation in N_2-seeded Discharges

Research of ammonia formation in N_2-seeded discharges has been carried out by residual gas analysis of N_2-seeding experiments. The RGA recordings of more than 2000 discharges have been analysed, among which are 111 N_2-seeded. In a mixed D-H system, the mass peaks of the possible isotope configurations of ammonia overlap with those of methane and water, most promiment impurities detected by RGA, in the 15-20 AMU range. In order to resolve all three molecules, a new model has been developed. The so found linear dependency between N_2 and ammonia supports the hyptothesis that the main driver of the ammonia production is the flow of nitrogen radicals from the plasma to the plasma facing surfaces.

Toroidal Structure of ELM Associated Magnetic Fluctuations

So called edge localized modes (ELMs) are accompanied by mode-like magnetohydrodynamic (MHD) activities with measurable toroidal mode numbers. These ELM associated fluctuations might play an important role for the onset of the ELM crash, as their structure seems to be conserved during the crash. On the ASDEX Upgrade tokamak these fluctuations were characterized in terms of their mode numbers magnetic pick-up coils. It was found that typical toroidal mode numbers of these modes are in the range of $n = -1 \ldots -12$, where the minus sign means that they rotate in the electron diamagnetic direction. On the JET tokamak very similar results were found. Figure 3 shows that slightly smaller ($n \geq -1 \ldots -16$) values, also rotating in electron drift direction.

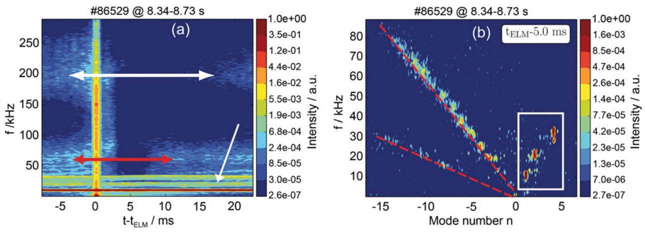

Figure 3: ELM synchronized (a) frequency histogram and (b) mode number histogram of the time window 5 ms before the ELM crash.

Inter-ELM Pedestal Recovery on JET

Understanding the post-ELM pedestal recovery on JET has recently been extended to edge current density modelling. The bootstrap current was calculated using the Sauter formula and the measured temperature and density profiles at different points during the ELM cycle. To determine the Ohmic portion of the edge current density, a time dependent current diffusion model was implemented, using the measured loop voltage as a starting assumption. As the profiles crashed at the ELM, the bootstrap current decreased, while the loop voltage increased sharply. Initial calculations have indicated that, while the Ohmic contribution to the pre-ELM edge current density is small (~10 %), it can make up 50 % of the total edge current density during the recovery phase. Current diffusion timescales were found to be in line with AUG findings at similar pedestal temperatures and densities.

Modelling of the Influence of SOL Conditions on LH Threshold

EDGE2D-EIRENE simulations were performed in order to understand a possible reason for lower H-mode power threshold in magnetic configurations with the outer strike points on the horizontal target (HT) compared to the vertical target (VT). Simulations revealed a strong difference in plasma density and temperature near the outer strike points in the two configurations, related to the difference in neutrals recycling pattern. This difference led to a formation of large positive radial electric field (E_r) in the near SOL in the HT configuration which was not seen in the VT configuration. With E_r being negative in the outer core, large E×B shear across the separatrix in the HT configuration may be responsible for a turbulence suppression in the vicinity of the separatrix, leading to lower H-mode power threshold.

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Stellarator Research

Wendelstein 7-X

Head: Prof. Dr. Thomas Klinger

1 Introduction

After 15 years of construction, Wendelstein 7-X started operation in December 2015. Despite the strong limitation of the injected energy and the unfavorable wall conditions, the first ten weeks of operation were a great success. More than 1000 experiments were conducted and about 30 diagnostic systems were put into operation. The device turned out to operate reliably and with surprisingly good performance. Wendelstein 7-X is now step-wise completed for high-power long-pulse operation capability.

In 2015 the organisation of the project "Wendelstein 7-X" (see figure 1) underwent major changes. The project was renamed "Wendelstein 7-X/C" ("C" stands for "completion") and streamlined to the future needs and the specific requirements arising from device completion and operation.
The project Wendelstein 7-X/C is a separate organisational structure besides the re-established five IPP scientific/technical divisions in Greifswald. The project Wendelstein 7-X/C consists of the following project divisions: "Project Coordination", "Design Engineering", "Assembly", and "In-vessel Components". Two staff offices "Safety" and "Quality Management" support the project. The interface to each of the three experimental physics divisions (E3, E4 and E5) and the operations division (OP) of IPP Greifswald is managed by a respective responsible officer. A project science council, in which all directors of IPP Greifswald are members, is in charge of defining the scientific goals of the machine and to set priorities. The responsibility and the budget of all diagnostic systems and the heating systems were transferred to the experimental divisions of IPP Greifswald. The responsibility and the budget for control and data acquisition systems and device operation (including the magnets, the vacuum systems, the cryo systems) were transferred to the operations division of IPP Greifswald. The remaining task of the project Wendelstein 7-X/C is the completion of the in-vessel component system, the installation of the remaining device periphery (cooling lines, cryo lines), and the installation of diagnostic and heating systems handed over by the responsible IPP divisions. This task requires that the project Wendelstein 7-X/C plays a role in the overall coordination of all activities related to the completion of the machine, including schedule and budget monitoring, quality management, change management, and design coordination. The organizational chart of the project Wendelstein 7-X/C is shown in figure 1.

The current status of the completion works on Wendelstein 7-X is described in chapter 2. The successful start of operation of Wendelstein 7-X in late 2015 was clearly a highlight: The experience with commissioning and first device operation is reported in chapter 3. The diagnostic systems, the heating systems and the first experimental results of the first operation phase of Wendelstein 7-X are discussed in chapters 4 to 6.

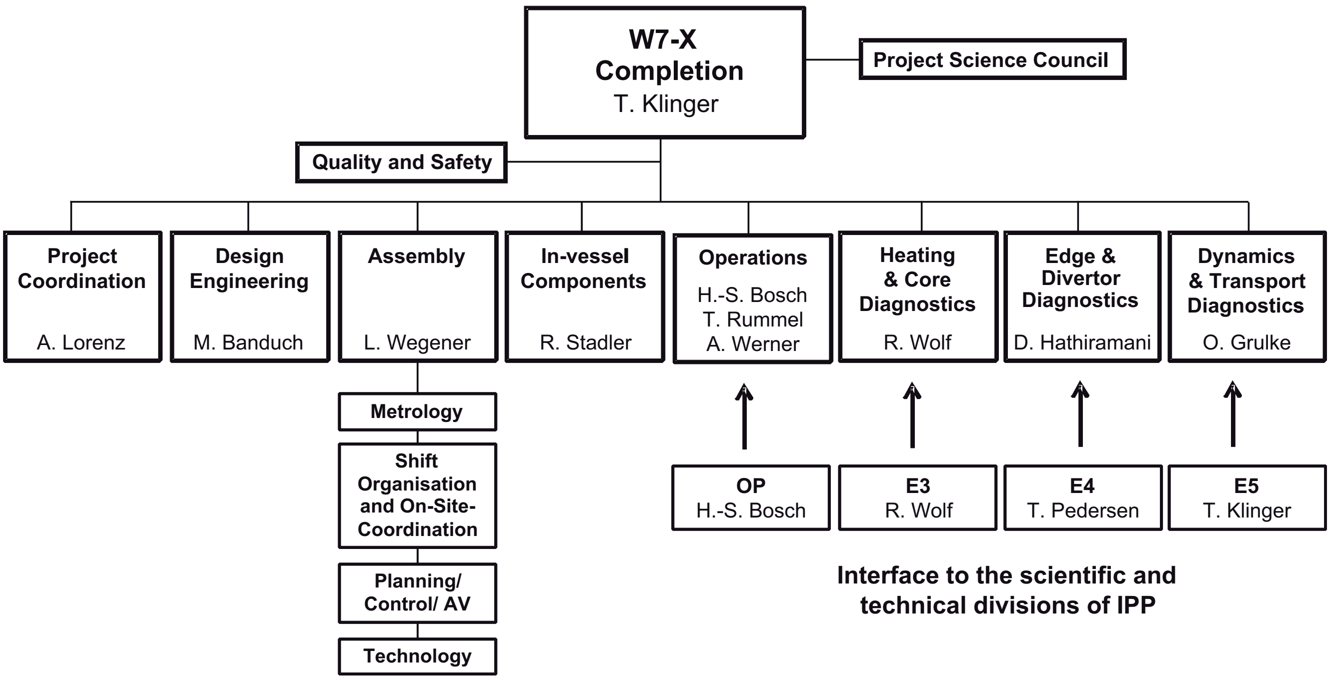

Figure 1: Organisational structure of Wendelstein 7-X project as of 01.09.2016.

2 Wendelstein 7-X Completion

2.1 Quality Management

The Quality Management (QM) staff unit reports to the directorate of IPP via the project director of Wendelstein 7-X, who has the direct responsibility for the maintenance and continually improvement of the QM system. The department organizes the QM system for the operation of Wendelstein 7-X and within the project Wendelstein 7-X/Completion and supports the supervision of external suppliers. It is responsible for quality assurance during all assembly, commissioning and operation phases of Wendelstein 7-X. The QM system certified since 2010 has been recertified by the TÜV NORD CERT in November 2015 for another period of three years.

2.2 Project Coordination

The role of the project division "Project Coordination" (PC) has been adapted to satisfy the needs of the project. It is responsible for the coordination of the financial planning, the control of the investment and operational expenditures and the time schedule monitoring of all activities within the project. With regard to the scientific and technical divisions, PC takes the role of a controlling and integration interface.

2.2.1 Planning

The division has been streamlined to a single structural unit with a head and five team members. The main project tasks have not changed. (1) monitoring and co-ordination of component delivery, (2) supports of responsible officers in the handling of external procurement contracts, (3) monitoring of assembly and commissioning schedules, (4) organizational aspects of the project and reporting to the external supervising bodies. An important new task is the central co-ordination of the design review process, reinforced and restructured in 2015 in view of the large amount of new systems to be integrated to the existing environment.

The integrated planning tool (IPT) is a MS Project based financial and time planning system. It remains being the routine planning, controlling, and financial reporting tool in the project. A major upgrade of the tool to MS Project 2013 has been launched in 2015, including an overhaul of the reporting portfolio. In addition a web based data base was initiated to provide an immediate overview on the current state of development and documentation of sub-projects ensuring the "single source of truth" principle. In its first version, the tool will be focused on converting the project specification data (components, media requirements etc.) from numerous distributed sources into one data base.

2.2.2 Schedule

After completion of OP1.1 in March 2016, the project schedule planning was concentrated on detailing the next operation phases OP1.2 and OP2. The operation phase OP1.2 is split up into two periods OP 1.2a and OP 1.2b to have access to the plasma vessel for additional installations (e.g. scraper element, ICRH antenna). The start of OP1.2a is foreseen in May 2017, the start of OP1.2b in March 2018 and the start of OP in April 2020. Keeping the schedule for the timely start of OP1.2 poses three main challenges:

a) Tests after OP1.1 have revealed a high voltage weakness of the non-planar coil circuits type 3 and 4. Immediate and careful investigations were launched. In the case that repair actions become necessary, the critical path of the project will be most likely affected.
b) The assembly planning of the in-vessel components has largely proven to be realistic and robust. After initial difficulties, which were mainly due to the high demands on the positioning of the uncooled divertor (TDU), the assembly cycle for the "series assembly" of 120 TDU modules has stabilized.
c) The Control and data acquisition (CoDaC) system has undergone a "bottom up" scheduling to meet the extended safety and operational requirements. This requires more manpower than initially foreseen and reinforcement is being implemented. On the other hand, a lot of diagnostic end systems are descoped, as they do not require steady state operation features before OP2.

Component and assembly preparation for OP 2 is also in full flow. The long-running HHF in-vessel components are in their final manufacturing and testing phase. Most importantly, the technically most challenging work packages for the HHF divertor are running according to schedule. The final assembly of the HHF divertor remains a big schedule risk. Recent experience from the installation of the TDU suggests that mounting the HHF divertor with its high demands on positioning accuracy (<200 microns), larger weight (70 kg per module) and huge amount of water pipe weld connections requires significant preparation and training. Thus, procured a 1:1 "mock-up" of a plasma vessel sector, will be including steel models of all important in-vessel components. The mock-up will be of great importance to further develop efficient assembly procedures. Current planning suggests that the project Wendelstein 7-X Completion is still on track to go into the important operation phase OP2 in spring of 2020.

2.3 Design Engineering

The project division "Design Engineering" provides design engineering solutions and the spatial planning for Wendelstein 7-X.

2.3.1 Work Organisation

In the year 2015 the former sub-division "Design and Configuration" was changed into the project division "Design Engineering". This was done in the course of the reorganization of the IPP Greifswald branch institute. In 2015, the goal of completing all design engineering tasks, as required for OP1.1, was successfully achieved. The majority of the tasks were design works for peripheral systems, like extending the media supplies, completing the cable tray system, extending the ensemble of platforms. In particular, the concerted action of completing the design and procurement of the cold water cooling system for OP1.1 was completed in time. In addition, numerous design engineering tasks concerning diagnostic systems for OP 1.1 were completed. The design engineering for further diagnostic systems for OP 1.2 has already made good progress. To track the priorities in the design works, a master table for all design engineering tasks was developed and is fully accessible by all Wendelstein 7-X staff. It updated on a regular basis and provides information about the task description, the current status and priority. A considerable increase of the design engineering capacity became necessary to cope with all high priority tasks in the list.

2.3.2 Spatial Planning

A fundamental principle for planning of construction space is the effort-risk-balancing. The principle of space allocations was changed over the past years: Previously it was tried to allocate space in the CAD-model to ensure that in future there will be absolutely no space conflicts. It was tried to allocate space for objects without even knowing their rough shape (the more uncertain the final shape was, the bigger the allocated space), partly without any supports or holders ("big flying boxes"). Initially this method was a reasonable approach to achieve a starting point for the planning. But with an increasingly crowded situation with more and more objects, the number of potential conflict points has strongly risen, causing a significant overhead in the actual design process. Another issue was that design engineers were not respecting the allocated space and their design was often sticking out. It became clear that a more pragmatic but reliable basis for space allocation is needed: It is not the aim to ensure that space conflicts are avoided by any means, but to identify the space for which it is evident that any penetrations cause problems. This space is called "essential space". This drastically reduces the effort for space allocations for the price of a slightly higher risk of space conflicts discovered later.

2.3.3 Design Engineering Tools

Of great importance for efficient design engineering is software development support. The development of comprehensive tools forms the fundamental basis for the daily work, e.g. a tool to generate a digital mock up of the torus hall and tools to reduce manual work while improving the quality of the results.

2.4 Assembly

During the first half of 2015, the "Assembly" project division was busy with remaining peripheral work until the start of first machine operation (OP1.1). This concerns piping and cabling works, support of the installation of diagnostics and electron cyclotron resonance heating (ECRH) system and many "last minute" items. In parallel the completion works of the in-vessel components were performed together with the preparation of the calibration system for the neutron counters. Furthermore, numerous repairs and improvements had to be performed, resulting from finding from the local commissioning of sub-systems. The boundary conditions for assembly works during this period were complex since local commissioning and assembly ran in parallel. The organisational safety-measures for works in the experimental hall were adapted to the increased safety risks. Another organisational challenge was the stepwise decrease of the assembly staff and the formation of a small core group. This group had to be adapted to future tasks in the operation division and to project completion works. The first systems to be commissioned were the vacuum systems and the cooling circuits. For the actual commissioning process, so-called "Commissioning Assurance Templates" (CATs) were developed, based on the format of the well-established "Quality Assurance and Assembly Plans" (QAAP) that have been used for structuring, controlling and monitoring of the various assembly processes. Consequently, CATs became the basis for the controlling and documentation of the commissioning of all sub-systems and components. As of the second half of 2015, the main task of the project division was the preparation of the next assembly phase (CP1.2a) starting March 2016.

2.4.1 In-vessel Components for CP1.2a

For the installation of the test divertor unit (TDU) modules, new processes and equipment were required. From the experiences of the first assembly phase it was already clear that the theoretically required assembly tolerances would be a severe challenge. On the other hand, the safe handling of the up to 40 kg heavy modules, the extremely limited space in the vessel and the protection against mechanical damages required special lifting tools. In addition, the development had already to take into account the requirements for the water-cooled high heat-flux (HHF) divertor with twice as heavy parts, to be installed for the later operation phase OP 2. To resolve the issue, a new concept was worked out and contractors were requested to develop a prototype. With this approach, a wide span of industrial competence and competing concepts could be included in the development process in the most efficient way. Several specialized companies were asked to develop a prototype, but only two of them accepted the challenge. Only one of the two was able to deliver a functional prototype by March 2016. During the test of the prototype it became clear that modifications were necessary for each of the 14 individual TDU-positions to cope in particular with the spatial limitations. Hence, the prototype was further developed and now includes construction kits to cover all 14 positions. The procurement of several construction kits allow for assembly in all five modules in parallel.

The kits were delivered in August 2016 and they work as planned. Another issue is the as-assembled accuracy of TDU modules. Two effects have to be considered: One issue is the assembly accuracy of the installed TDU metal frames. The TDU frames carry three to four TDU modules each. The achievable frame-accuracy of <1.5 mm stands in contrast to the theoretically desired one. A second issue are small steps between TDU modules. They were accepted as long as they comply with the proper magnetic field direction. Only one TDU module per divertor has to be adjusted via customized fastening elements to compensate for the as-built TDU frame position. The third issue are steps within one module or between modules on one frame (up to 0.5 mm). Steps can occur after the replacement of carbon elements ("fingers"). There is hardly a reliable and practical way to fix these steps within acceptable costs and process times. At the moment strategies are being developed to cope with the occurred assembly tolerances. One strategy is to accept the situation as it is and to learn during the operation phase OP 1.2 about the actual requirements. Based on the lessons learned, the later installation of the HHF divertor requires further developed assembly technologies and equipment as well as a 1:1 mockup of a full module of the plasma vessel. The simple transfer and extension of the assembly process from the TDU to the HHF divertor is not sufficient. The assembly of baffle modules (that follows immediately after the TDU assembly sequence) was further refined. An issue is again the achievable accuracy: The requested tolerance of <1.5 mm cannot be guaranteed. In single cases not less than 2.5 mm can be achieved. The main reason is that, due to space limitations, the functions "hold" and "align" are combined in one element. The same baffles will be used for the HHF divertor but it may become necessary to develop and procure a new fastening system. During assembly tests of the baffles it was realized that a number of them cannot be connected to the cooling circuits without a redesign of their terminations. Though that is only relevant for the later operation phase OP 2, the redesign and modification by industrial partners has been started immediately. Carbon tiles to cover the heat shields and baffles are installed during CP1.2a. Measurements have revealed that the as-built position of heat shields and panels can lead to clashes between tiles. It was thus decided to establish a dedicated tile-customization process: Special high-speed machining-equipment was procured and set up in the machine shop. The as-built contour of the heat shields and baffles is recorded with a small and easy-to-use scanning device. Its use by the assembly personnel runs without any complications. The surface to be scanned is subdivided into sections that are compiled in the computer with the help of reference marks that cover all components. The processing of the scanned data to obtain customized CAD-models of the tiles is outsourced to an external company. The scanning, data processing, and machining of the tiles runs as planned and the assembly sequence of tiles has been adapted to the availability and capacity of the external partners. All tiles need to be cleaned after machining. The original idea to clean the tiles in an ultrasonic bath und to dry them in a 300 °C oven was abandoned since outgassing tests have shown a too high level of remnants, probably from the air in the machine shop. These remnants show sufficient desorbtion at temperatures >500 °C and suitable ovens are now used for baking to ensure the required cleanliness. Unexpectedly, the scans have revealed severely deformed heat shields in areas that are characterized by strongly restricted space for equipment and assembly workers. The heat shields are obviously not robust enough to withstand accidental pushes during running in-vessel works. The affected parts were manually corrected and supported against the vessel wall. Additional protection measures are needed for the TDU since the installation of the target modules is impossible without stepping on them. The deadweight of the component and a man on a divertor module must be distributed across its entire surface. The problem is that these modules are not flat and not purely horizontal. A possible protection must be easily to handle and lightweight, it must not slip, no abrasion, no sharp edges, and it must be 3D-shaped according to the CAD contours but compatible to as-built deviations in the millimeter range. A new campaign was launched in autumn 2015 to resolve this known issue. Industrial partners that are specialized to the fabrication of mattresses were asked for support. Specialized cushions have been proposed, which have an inverted contour to the CAD contours that lie on the divertor surface. The right composite ensures that small deviations are compensated through a soft layer, whereas the full weight is distributed through a layer of large shore-hardness. A glued thin aluminum plate provides the best possible horizontal stepping surface for the worker. A foil-envelop prevents abrasion or slippage. This protection is lightweight and can be easily modified by the supplier. For OP 2 about 90 port liners are required to prevent overheating of weld seams and bellows. Also the overheating of the actual ports must be prevented (<80 °C) to avoid malfunctions in adjacent cryogenic components. Overheating can be caused through plasma radiation and ERCH stray radiation. Only 10 mm space is available for the port liners and the design must be strictly within this margin. At the plasma vessel side, port liners can have fairly exotic contours and the active cooling is quite challenging. A prototype program has been started and first samples will arrive at the end of 2016. More details are explained in section 2.5.6.

2.4.2 Periphery and Support for Diagnostics

About half of the assembly capacity is used for the installation of pipes, cables, and for the installation of diagnostics. The largest share is needed for the pipework of the extended cooling circuits and heating systems (NBI, ICRH). Further work packages are the support for the setup of titanium sublimation pumps (NBI), repair works (e.g. the plug-ins of control coils, ducts of the vacuum systems), general welding support, and the extension of platforms and support structures.

2.4.3 Technology, Planning and Organisation

About ten engineers are working in the technology group to develop the assembly procedures and the needed assembly equipment. Six of them are focused on in-vessel works including the embedded diagnostics. All assembly procedures are described in work instructions. Seven engineers are working on the planning and work preparation to keep the weekly planning and the four-week planning up to date. They also manage the monitoring and control of the daily work progress. Works on-site are performed in up to 88 h/week. The assembly sequences are controlled via QAAPs (see above). Several hundred QAAPs are being created and edited through the work preparation group in collaboration with the responsible officers, the technology engineers, and quality management. The recruitment of the needed assembly staff for CP1.2a was demanding since this phase lasts only one year. The planning for the further completion phases is updated on a regular basis to include the gained experiences. In summary, the device assembly has reached the planned progress in 2016. The completion date of the assembly phase CP1.2a is still unchanged, including the foreseen contingency. The cooperation with industrial partners in the fields of personnel and technology was fruitful and effective.

2.5 In-vessel Components

2.5.1 In-vessel Components

In parallel to the commissioning of the experiment for the first operational phase OP 1.1, the detailed design, manufacturing and testing of the in-vessel components needed for the next operation phase OP 2 was continued. Concerned are the divertor components (target, baffles, and toroidal closure plates), plasma vessel protection (panels and heat shields), control coils, cryo-pumps, port protections and special port liners for the different heating systems together with a complex system of cooling water supply lines inside the plasma vessel. The main components to be completed are the high heat flux (HHF) divertor, port protection liners and cryo-pumps.

2.5.2 Target Elements

For the long pulse, high power capability of Wendelstein 7-X, actively water cooled elements are needed, in particular for the HHF divertor. 890 HHF divertor target elements were manufactured by the company Plansee (Reutte). These elements consist of 8 mm thick carbon fibre reinforced composite (CFC) tiles, joined to a water-cooled CuCrZr heat sink. The elements are specified to withstand steady state power fluxes of up to 10 MW/m^2 and to operate with 12 MW/m^2 for a reduced number of cycles. During the incoming inspection, some target elements (of type 5) did not pass the leak testing performed in a vacuum oven with pressurized He. The leak was always found in the area of the water connection and there more precisely in the interface between the Ni adapter and the steel pipe of the water connector. After analysis of the possible failure causes and investigation of different repair possibilities, it was decided to develop in close cooperation with the company Galvano-T (Windeck) a Cu-coating to seal off the leak. The coating will be applied to all target elements of types 1, 2 and 5 as a risk mitigation measure. The serial coating has started at the end of the year 2015.

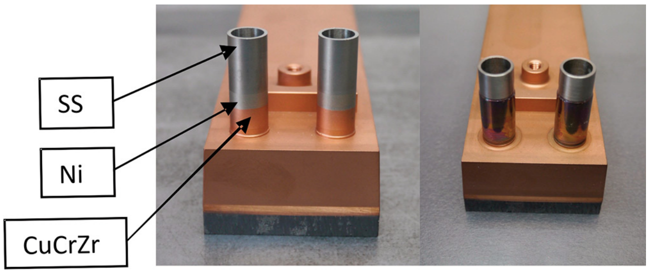

Figure 2: The connectors of TE 5 before and after coating.

2.5.3 Target Modules

The Wendelstein 7-X divertor consists of three main HHF areas: the vertical and main horizontal targets and the “high iota tail”. The targets are built from target modules, which are sets of mechanically and hydraulically connected target elements (varying from 6 to 12 target elements per target module). The “high iota tail” is made of the three modules TM7h, TM8h and TM9h and was completed in 2014. The manufacturing of the target modules is mainly done at the IPP workshops of the ITZ (“Integriertes Technik Zentrum”) in Garching.

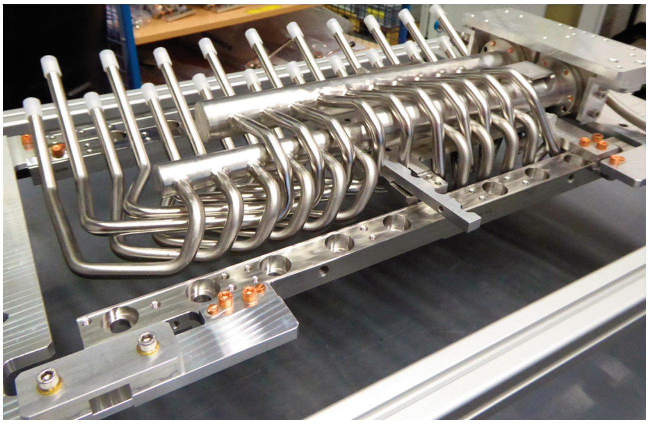

Figure 3: Inlet and outlet manifolds welded to pipes of TM3v.

The production of the vertical target area, with three modules TM1v, TM2v and TM3v has started. The water manifolds were delivered by the company Dockweiler (Neustadt-Glewe).

In 2015 different jigs for intermediate or final assembly were manufactured as well as the first parts of the support mechanical structure. Furthermore, the pipe welding of the manifolds was started. The vertical target modules will be connected to the water supply circuits via caged bellows that provide the required flexibility for installation and adjustment. Three different types of bellows were comprehensively tested. The final selection and procurement of the serial parts is foreseen in the year 2016.
The detailed design of the main horizontal target, with the four modules TMh1 to TMh4, was continued. Manufacturing of the first jigs and pipes for the water manifolds has started in 2015.

2.5.4 Test Divertor Unit (TDU)

The test divertor unit (TDU) is is an intertially cooled divertor solution to be used during OP1.2. It has exactly the same geometry as the HHF divertor but uses graphite tiles instead of water-cooled HHF target elements. The TDU is already in the assembly process for operation in the phase OP 1.2.

2.5.5 Baffle Modules

The manufacture of the main baffle modules was completed in 2013 and final testing was successfully performed in 2014. Due to the closure of some plasma vessel ports and changes in the Rogowsky coils, eight new baffle modules needed to be manufactured. These new baffles were finished in 2015.

2.5.6 Wall and Port Protection

Other in-vessel components are double walled stainless steel panels (covering approx 70 m^2 of the plasma vessel) and heat shields (covering approx 50 m^2), consisting of water cooled copper plates clad with graphite tiles (similar to the baffles). The assembly of these components (without graphite tiles) was completed in 2014. In the area of the remote ECRH-launchers, new steel panels with special cutouts are required. They were contracted to the company MDT (Deggendorf) and production started in 2015. The installation of these panels is planned for OP1.2a. For OP1.1, graphite tiles were procured for selected heat shields to form a plasma limiter and to protect the ECRH beam dump area. Four modified panels for two ports of the ECRH system were delivered and assembled before OP1.1. For the port liners, companies were asked to offer manufacturing concepts that match the specified mechanical and functional parameters, including calculations, analyses, and quality tests. In a second step, three companies were contracted to manufacture three prototypes each, based on the most promising concepts. The prototype are planned to be delivered in the beginning of 2017. Comprehensive tests are planned, including heat leak-tests and the measurement of the thermal behavior. A special element at port liners is the so-called “gap closure” that protects the weld seam between port and the plasma vessel. It requires a complex design and installation process due to its curved shape and the bad accessibility behind the in-vessel panels. Fins made of copper-plated steel stripes, welded to the actively cooled port liner provide the needed functionality for both the heat conductivity and mountability. Prototypes of this design were successfully made and assembled. It is planned to procure the gap-closure in conjunction with the port liner from the same manufacturers. The total heat load on ports and port welds have been updated and refined in 2016. As a result, in addition to the port liners, further protections at port bellows are necessary. First solutions are designed at the moment. The technical specification for the procurement of all port liners will be drafted in autumn. Its call for tender is planned for spring 2017. During the design of the vertical modules of the HHF divertor, it became clear that some previously designed and procured components surrounding the modules would have to be changed to avoid collisions with neighbouring components, e.g. the pumping gap panels, designed to protect the plasma vessel between the main horizontal and vertical targets. The design for these panels was detailed and start of the procurement process is in 2016.

2.5.7 Cryo-pumps

The in-vessel cryo-pumps, located behind the main horizontal HHF divertor target modules, have been designed and partly manufactured. Since the cryo-pumps will not be installed until the operation phase OP 2, manufacturing was stopped in 2008. The activities were restarted in 2014. Each of the ten identical cryopumps is divided into two divertor cryopump Units (DCUs).

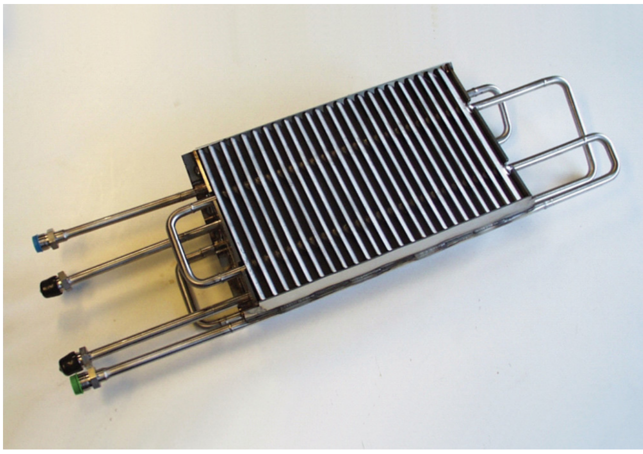

Figure 4: DCU2.

Figure 4 shows the completed DCU 2 with the LN_2-chevrons, He-panels and reflectors. Temporary pipe connections were welded to allow testing in a vacuum oven with pressurized He. Due to a re-assessment of the ECRH stray radiation loads expected for OP 2, the possible impact on the present design has to be evaluated.

The manufacturing of the vacuum feed-throughs to provide the cryo pumps with LN_2 and ScHe is done in-house by the local mechanical shop in Greifswald. The manufacturing of the cryo-shield that thermally protects the cryo feed lines has to be done by an external company. The feasibility of a weld procedure was demonstrated with test pieces, in close cooperation with the company PTR Strahltechnik (Langenselbold) figure 5. The procurement is planned for the end of 2016 in order to have all parts available for the start of internal manufacturing in 2017.

Figure 5: Test specimen of cryo shields.

2.5.8 Control Coils

The control coils (in-vessel coils control locally the magnetic field in the divertor region) were assembled in 2014. Commissioning of the coils was performed in 2015. One leak was found at the interface to the water feed-through. The coil was not filled with water and the repair was done during CP1.2.

2.5.9 Plug-ins

The in-vessel plug-ins are used to deliver water and, in some cases, diagnostic cabling from the outside of the machine to the inside of the vessel through the supply ports. The plug-ins consist of a set of tubes, welded onto a flange. All 80 plug-ins have been manufactured and assembled during 2014.

2.5.10 Water Supply Lines inside the Plasma Vessel

The cooling supply lines of the in-vessel components run from the plug-ins, via a complicated system of manifolds and pipes, to the various components via flanges (in total 308 cooling circuits). The pipework for the panels and the heat shields was completely welded before OP 1.1, but not all of them are filled with water until OP 2. Those panel circuits not filled with water are filled with inert gas to provide some thermal conduction to avoid hot spots.

3 Operations

3.1 Device Operation

3.1.1 Physics Operation

Preparation for plasma operation started with baking of plasma vessel and ports in August 2015 after the successful commissioning of the superconducting magnet system. In addition to vessel and ports, the ten control coils inside the vessel had to be actively heated as well in order to avoid intolerable thermal stress in the winding packs. All other in-vessel components were passively heated by heat conduction and radiation from the hot components. A few temperature sensitive diagnostic components in the plasma vessel and immersion tubes were cooled by water or pressurized air. The temperature was ramped up within two days. Intermediate plateaus at 80 and 120 °C, respectively, were kept overnight to allow for temperature equilibration and for safety reasons. The final 150 °C plateau was kept for seven days, followed by two days ramp down. The thermo-mechanical behaviour was monitored for plasma vessel, ports and vessel supports by thermocouples, displacement sensors and strain gauges and was in good agreement with the predictions from the global cryostat model. Finally, the removal of volatile impurities by baking effectively reduced the residual pressure in the vessel by a factor of ≈10 down to $2 \cdot 10^{-8}$ mbar. The glow discharge cleaning system for further improvement of the first wall conditions was not yet operational at this time. The systems required for plasma production and sustainment, i.e. gas injection and electron cyclotron resonance heating (ECRH), were separately commissioned using their local control systems. Most important for safe ECRH operation was the validation of the interlock which immediately stops the heating pulse in case that the µ-waves are not sufficiently absorbed by the plasma. The respective interlock signal is provided by so called *sniffer probes* which detect non-absorbed µ-wave power. For plasma operation, the required active systems were subordinated to the central segment control system, which provides their necessary synchronization. The monitoring and initial assessment of plasma performance in-between discharges was performed with the time-traces of the gas injection flow rate, launched ECRH power, sniffer probe signals, video diagnostics with eight cameras covering nearly the complete torus, the line integrated plasma density along a central chord, electron temperature profiles from ECE-diagnostics and pressure gauges providing the neutral gas pressure at the plasma edge. Operation started with Helium for reasons of safety. A short 20 ms Helium gas puff via a piezo valve provided the gas target for ECRH, which was launched 100 ms after the gas puff by two gyrotrons with 500 kW each. Break down was successful in the first attempt but ECRH absorption was lost after 10 ms, probably due to the influx of impurities still being adsorbed on the plasma facing components. By repeated application of ECRH conditioning cycles the gas load of the walls was reduced such that the absorption phase could be extended to 50 ms. Such a conditioning cycles is a sequence of up to 20 consecutive short discharges with up to 3 MW ECRH power, 50 ms pulse length and 30 s dwell time between the pulses to allow for pumping. Further improvement was achieved with the availability of the glow discharge conditioning (GDC) system.

After the first application of a He glow discharge with 15 min duration the absorption phase increased to about 100 ms, but degraded steadily in the following plasma discharges. Since GDC cannot be applied in the presence of a magnetic field, it had to be performed in the evening or in the morning when the field was not activated. Therefore, GDC was effective only for the first discharges of a day. Plasma performance improved substantially after with the changeover to H operation. Additionally, the length of glow discharge conditioning was extended up to 40 min, providing excellent conditions for the early discharges of a day. ECRH conditioning with Helium was rather effective to remove H from the walls. It was even sufficient to re-establish low recycling conditions by performing a single intermediate ECRH plasma discharge in He (recovery discharge). Already in the first H-plasmas the absorption phase reached 200 ms with 2 MW heating power but the discharges were still limited by an uncontrollable density increase which provoked a collapse. This improved rapidly with proceeding operation such that with moderate heating power of 0.6 MW a pulse length of 6 s at constant line-averaged density of $7 \cdot 10^{18}$ m^{-3} was possible.

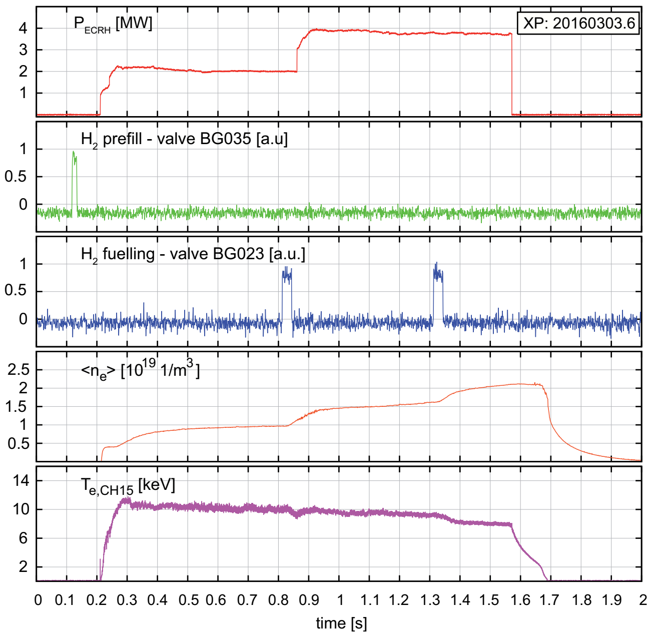

Figure 6: Discharge with density three plateaus adjusted by Hydrogen gas puffs.

Long discharges are of particular importance for the project Wendelstein 7-X, since the time scales for the development of internal plasma currents can be in the order of some seconds. Full density control was not achieved with the present limiter configuration even with Hydrogen. The effective recycling coefficient was limited to R_{eff}=1, i.e. without external gas fuelling the density stayed constant. However, with controlled gas puffing the density could be adjusted to higher levels, see figure 6.

3.1.2 Device Safety

Based on the safety analyses and dedicated risk assessments, protection measures for risk mitigation have been defined and implemented in order to ensure a safe design (redundancy, diversity, overpressure protection – eight projects are entirely dedicated to safety) as well as an appropriate level of functional safety (central safety systems, safety instrumented functions). According to the norm IEC 61511, the requirements on safety instrumented functions (SIF) were derived from the safety analysis for the main device as well as from the risk assessments of all associated components and compiled in the safety requirements specification (SRS), which contains 27 personnel related and nine device safety related SIFs for the operation phase OP 1.1. Residual risks have been met by additional organizational safety procedures (access control, safety briefings, operation permits). During the final phase of the assembly extended commissioning work was performed in parallel. During magnetic field operation of super-conducting coils, personnel were not allowed to stay in the torus hall (incl. the second basement). However, with ongoing vacuum and cryo supplies, assembly installation and service works including function tests of inactive components inside the torus hall and the cryostat was carried out. Potential safety risks which may exist when these activities are performed and the consequently required safety measures have been analysed and implemented. Special attention has been paid to hazards which may result from loss of vacuum, leaking of helium and fire incidents inside the torus hall. From pumping down the cryostat (July 2014), cooling down (Feb. 2015), and performing the magnetic field tests up to the plasma preparation phase (incl. pumping down the plasma vessel, baking, glow discharge cleaning) all systems together with the central safety control system demonstrated a high availability. Technical failures, not unusual during a commissioning phase, could be fixed without considerable impact on the project schedule. No quench occurred, but two unplanned short discharges of the magnet system due to an outage of the general (external) power supply (for about 300 ms) during magnetic field tests and due to a fast ramp-down of the trim coils triggering the QD system by inducing a voltage in the nearby located planar coils. No failure of the central safety system has been observed during OP 1.1. Stable operation of all required technical systems during the commissioning phase and plasma experiments 2014-2016 has been achieved which resulted in a 94% operational availability.

3.2 Magnets and Cryosystems

3.2.1 Magnet System

Wendelstein 7-X has a superconducting magnet system consisting of 50 non-planar and 20 planar coils. Seven electrical circuits with 10 coils each (connected in series) allow one to run individual currents in the seven coil types. The 10 coils of the same coil type are connected in series by superconducting bus bars.

In total 121 bus bars connect the coils in the seven circuits. The bus system uses the Wendelstein 7-X superconductor with slightly rounded jacket edges to allow bending in all directions. Specially developed high temperature superconducting current leads (CL) feed the current into the cryostat by bridging the temperature gradient from room temperature down to the 4 K level. Each of the seven coil circuits has its own power supply to provide individual currents. The construction of the power supplies with the integrated magnet protection system is similar for all seven circuits. The quench detection system (QDS) monitors the superconducting components continuously regarding the development of voltages which indicates the loss of superconductivity. By comparison of two section voltages (one double layer is one section) in one quench detection unit, the QDS is able to eliminate electromagnetic interferences and to detect voltage differences between the two sections. In addition to the superconducting coils, normal conducting coils, the so called trim coils are mounted on the outer cryostat wall, one coil per each of the five Wendelstein 7-X modules. Their commissioning was already completed in 2014. The superconducting magnet system commissioning was successfully performed during the period between April and July 2015. The commissioning started with tests on the single coil type circuits and was continued with tests on the complete system. The magnet commissioning was accompanied by an online monitoring and evaluation of the temperatures and of the mechanical sensors to guarantee a proper function of all involved components. Subsequent to the magnet system commissioning, the magnetic flux surface measurements started with an extensive survey program checking the accuracy of the magnetic field. The commissioning strategy had to respond to the following challenges: the power supplies operate first time with full inductive load of 1 H for non-planar coil circuits and 0.4 H for planar coil circuits respectively, the superconducting bus bars with the related joint connections carry for the first time an electrical current and the QDS needs a balancing process for the internal measurement bridge with current ramps while the QDS is not functional. The magnet commissioning was structured into three main phases. The 1st phase was necessary to bring the QDS into operation by a balancing program with 500 A pulses in each coil circuit. During this phase the QDS was not active. Analysis had shown that the superconducting parts e.g. coils or the bus system could carry the relative low current of 500 a even in normal conductive state. The functionality of the single coil circuit components was tested in the 2nd commissioning phase while the current levels were stepwise increased. To avoid overloading and excessive displacements for NPC 1 and 5 – the current was limited to 10 kA for these types. The 2nd phase contains four current levels with a similar test procedure. The magnet protection system was tested at each current level during fast discharges with increased energy dissipation into the dump resistor.

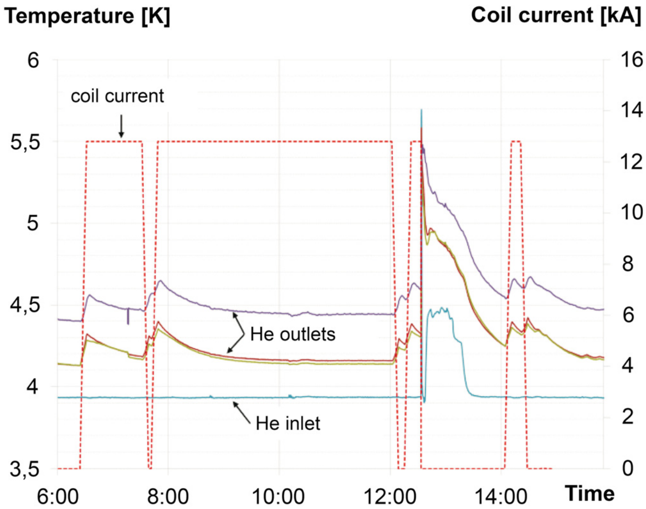

Figure 7. Temperature behavior during current tests of non-planar coil type 4 on 16th June 2015, the temperatures are the mean values of the 10 series connected coils.

Figure 7 presents the temperature behaviour of the NPC4 coil types during current ramps and during plateau phases. AC losses during the current ramps of 30 A/s increase the temperature on the winding outlet of about 0.2 K. A fast discharge with ramps of maximum 5000 A/s increases the outlet temperature of about 1.3 K while the casing temperature increases up to 10 K. The ohmic heating losses in the coils can be derived from the temperature differences between inlet and outlet by comparison of phases with and without current. As result of this comparison, the ohmic heating leads to a 38 mK temperature increase in average implying 2.9 nOhm resistance in the coil during plateau phase. This value is determined by the five internal normal conductive joints connecting the six double layers in series. The 2.9 nOhm value is well below the specified joint resistance. The first plasma operation phase of Wendelstein 7-X requires a magnetic field of 2.5T on the plasma axis. To provide the magnetic field the five non planar coil (NPC) circuits were be operated with 12.8 kA and the two planar coil (PC) circuits with 5 kA respectively. The objective of the commissioning phase was: 1st Stepwise commissioning of the magnet system components with full functionality for steady state operation by observing at least 1 K safety margin for superconducting components; 2nd Adjustment of cooling flow in the superconducting parts especially the current leads and 3rd Continuous monitoring of the mechanical sensors, of the He flow and the component temperatures with benchmarking of the data with the Finite Element (FE) models or with specified values. In the 3rd commissioning phase the test sequence from the previous phase was repeated with all seven coil circuits in parallel operation. Finally the current in the PC was varied between 0 kA and 5 kA while the NPC carried 12.8 kA.

While the NPC were operated at 12.8 kA and the planar coils with 5 kA respectively, a fast discharge was initiated to check the magnet protection system. The magnet system stores 430 MJ at this stage, during a fast discharge the energy is conducted into dump resistors for commutation into heat. During a fast discharge starting at the 2.5 T level a high voltage up to 2.1 kV against ground potential occurs in the non-planar coil circuits, respectively 600 V in the planar coil circuits. The values are in good agreement with the expectations.

The first plasma operation phase OP 1.1 required a magnetic field of 2.5 T at the area where the ECRH waves should be absorbed from the plasma. To provide the magnetic field, the magnet system was operated with the magnetic field configuration "OP 1.1 Limiter" which was defined with 12.8 kA in the five NPC circuits and 5 kA in the two PC circuits respectively. To shift the ECRH resonance into the plasma centre, the coil currents were fine tuned in small steps. Finally the configuration with 12.37 kA in the NPC circuits and 4.82 kA in the PC circuits was found to be the optimum regarding the absorption of the ECRH waves. The superconducting magnets were energized during OP 1.1 for about 183 hours spread over 31 operation days. The magnet system was energized 35 times up to 2.5 T level during OP 1.1. The availability of the magnet system was approximately 94%. Parallel to the superconducting coils the five normally conducting trim coils were operated during the operation phase OP 1.1 at different current levels according the physics program.

3.2.2 Cryogenics

3.2.2.1 Preparation of the Cryogenic System

The commissioning of the cryogenic system was a part of the commissioning of Wendelstein 7-X. It comprises the preparation of the refrigerator, the supply lines to the cryostat, the piping inside the cryostat and the quench gas exhaust system. The following procedure was applied: (a) Verification that assembly and testing of all involved systems had been finished, (b) filling of individual pipe sectors with nitrogen gas up to 3 bars and rapid expansion into the surroundings for cleaning purpose, (c) evacuation to 30 mbar and filling of cooling circuits with 1 bar He (repeated three times) (d) further cleaning of the He circuits with the dryer and the 80 K adsorber of the He refrigerator (removal of water and N2) (e) checking of He leak rates of all cooling circuits inside the cryostat. Overall a helium leak- rate of about $3 \cdot 10^{-5}$ mbar l/s was measured.

3.2.2.2 Cool down

The cool down of the thermal shield and the magnet system was done in parallel with the cool down of the refrigerator and started from room temperature. At the beginning the insulation vacuum was better than $2 \cdot 10^{-4}$ mbar. The thermal shield was cooled in parallel with the cold structures down to the dedicated shield inlet temperature of 50 K. The cool down from room temperature to about 6 K was achieved within 24 days. The allowed maximum temperature difference between cooled components and the helium inlet temperature was defined to 40 K. This criteria limited the actual cold down rate of the heavy steel structures below 1 K/h. Visual checks of the outside of the cryostat, pipes and wire feed-through, safety valves and transfer lines didn't show any ice or water condensation. There weren't any noticeable vibrations or noise by an instable helium flow. Measured displacements were within the predicted range. The cryostat pressure inside the cryostat dropped down to 10^{-7} mbar simultaneously with the cool down of the coils.

3.2.2.3 Operation

The required operation modes of the cryo plant were tested after the cool down. First the short standby mode and then the standard operation mode were checked. The controllers of the He refrigerator were adjusted. Cold pumps and a cold compressor could be operated and allowed a stable cooling of the coils, bus bars, and the current leads. The defined cooling parameters with respect to He inlet temperatures and mass flow rates were achieved (3.9 K and 500 g/s for the coils and the support structures). The heat loads on cold components and the conductor outlet temperatures were within the specification. The shield inlet temperature was around 50 K. The heat load on the shield matched the design criteria of the thermal insulation (6 W/m^2 between ambient temperature and 80 K surface). The achieved cooling parameters and conductor outlet temperatures allowed the charging of the magnet system with currents up to 12.8 kA for the non planar coils. When a constant current is operated in the magnet system, the cooling conditions could be kept stable over hours.

3.2.2.4 Warm up

The warm up of the cryostat was done with the same criteria as for the cool down. It was started March 2016 and took five weeks. The average warm up rate was 0.6 k/h. During easter holidays the inlet temperature was constant at around 96 K. The warm up was very smooth as the operators of the helium refrigerator had collected experience in running the plant. The cryostat was kept cold over one year without major problems. The helium refrigerator was quiet reliable. Only nine incidents resulted in a trip of the whole plant. Problems could be solved in most cases within 1-2 days. Problems in the cooling water system and in external supply systems were the major reason for the trips.

3.2.3 Vacuum Technology

Until 2015 all three vacuum systems (interspace-vacuum, cryostat, plasma vessel) were completed and commissioned for the operation phase OP 1.1. A combination of five turbo molecular pumps, roots and rotary pumps is used to evacuate the cryostat volume (380 m^3 with inner surface of approx. 100,000 m^2) which achieves at 4 K interior temperature a pressure of to $3 \cdot 10^{-7}$ mbar at a remaining outgassing and leak-rate of approx. 10^{-2} mbar·l/sec. 30 turbomolecular pumps with their pre-vacuum system (10 roots and rotary pumps each) evacuate the 130 m^3 plasma vessel to $1.5 \cdot 10^{-8}$ mbar at a remaining outgassing and leak-rate of approx. 10^{-4} mbar·l/sec. The values were achieved after baking and after several conditioning cycles (glow discharges in He). As expected many weeks were necessary to look for leaks at the closed vessels with their hundreds of gasket-sealed closures and weld seams. Larger, systematic leaks were identified and repaired at the ten pump-ducts of the plasma vessel, shortly before the commissioning should start. Two of these were fixed with a welded workaround. Their comprehensive repair was shifted to the completion phase after OP 1.1. Comprehensive resources were allotted to these repairs to not jeopardize the start of the commissioning. Some other leaks at the outer area of the machine were identified acoustically (also ultrasonic). In the high-vacuum range the usual mass spectrometer and local helium injection were used for the leak detection. Not correctly tightened bolts at flange gaskets (mainly at belatedly installed components) and few failures at welds (mainly because of design issues) were the reasons and could be repaired swiftly. It must be noted that at the outer vessel only few leak tests were performed during the assembly mainly at badly accessible spots to save time. Instead, a strict visual quality control was used. That strategy relieved the critical path of the project with an only moderate increase of the risk. None of the leaks did hamper the start of the operation phase. A large work-package was the setup of the gas inlet system. The installation of the associated piping, the valve-boxes and the sensors went on as planned. Several assembly resources were temporarily allotted to this task. A real bottleneck and problem was the setup of the control system for the above systems. Both the procurement and installation of the electrical equipment as well as the programming of the control units had to be made on a week-by-week basis, since a functional specification that reflected the complexity of these systems was not available yet. That aggravated an effective work in an anyway quite tensioned resource situation. Some planned and not safety-relevant functions were skipped from the task list at that time, and their implementation was postponed to later completion phases. These functions were performed through manual control during the first operational phase. In operation the above systems worked as planned with the limitation of some control functions. The gas inlet system showed the necessity for improvements and modifications to ensure a precise and swift plasma creation. Three glow discharge electrodes failed and had to be put out of operation. Some larger valves separating diagnostics from the plasma vessel were overheated during the baking cycle of the machine. Misplaced sensors of the electrical port-heating system at the outside port area were the reason. The data transfer-system worked instable and failed some times. In 2016, after the first operation phase, the repairs and improvements are being realized as described before. A maintenance program has been structured and is successively executed. The gas-inlet system is being upgraded. As an additional project, the design and procurement of a waste-gas treatment system (pyrolysis of Diborane) has started. The system is situated in two of the ten pump units of the plasma vacuum system. It consists of one electrical heater each and of one additional chemical absorber for the redundancy. The project runs as planned and shall be accomplished in spring 2017. The control systems are extended and further developed based on both a requirement specification and the associated functional specification.

3.2.4 Engineering

In 2015 the group "Engineering" has concentrated on the commissioning of the about 800 mechanical sensors (strain gauges, distance and contact sensors) at the magnet and the cryostat systems. In addition the existing finite element models (FEM) were benchmarked with the signals and their handling were refined and optimized. Expected measurement values were pre-calculated with these numerical models according to the planned commissioning and operational regimes. Important additional information has been collected during individual coil group commissioning due to different loading pattern in comparison with simultaneous ramp-up of all coils. During operation engineering monitored the mechanical behaviour of the above systems in parallel with temperature and current levels and appraised the measured values. That was a part of the daily decision making. The software for the handling, the analysis and the monitoring of data was further optimized and improved in parallel. Though the measured values at 2.5 T operation lay below the technical limits their simple scaling to the 3 T operation might mean a local overloading at some supports of the magnet system. However, inaccuracies in the measurements could also explain these apparent results. Further analyses are made to interpret the measured values more precisely. Since 2016, after the first operation phase, engineering carries out FEM-calculations to support the design of new components mainly for the later HHF phase (e.g. port liner, cryo pumps). Maintenance and expansion of the existing FEM and the pre-calculation of future operational regimes are made in addition.

3.3 CoDaC

3.3.1 General

The group "Control, Data Acquisition and Communication" (CoDaC) enabled successfully the Wendelstein 7-X commissioning and experiment operation with the of the control and data acquisition systems. This comprises the central safety system (cSS), the central operational management (cOPM), the plasma control based on the segment concept, the continuous data acquisition as well as the control room with virtualized desktop computers and monitoring on the video wall. Many systems have been setup and engaged for operation, e.g. the central valve control, gas inlet, interferometry, flux surface measurements, ECE, HEXOS, Magnetics, Video and support for partly integrated diagnostics.

3.3.2 Control and Data Acquisition

The central safety control system has been set up according to the appropriate engineering standards (EN/ISO 61511), which was a precondition for the Wendelstein 7-X operation permit. This includes the setup of a fail-safe redundant Simatic S7-400 CPU, the operator panels and many interface cubicles for the safety signal exchange with auxiliary components like ECRH, magnet power supply, cryo systems and many more as well as diagnostics. In particular the door locking system for radiation protection zones and the radiation protection system were of high importance and subject to a formal development process, based on modern modeling techniques (SysML), validation procedures and several inspections by the German “Technischer Überwachungsverein” (TÜV, association for technical Inspections). The central operational management has been engaged for operation including status displays on infrastructure, process survey functions like the critical current of the superconducting magnets and the data dispatching function, which is the basis of the distributed control system regarding communication and process variable exchange. The segment plasma control system has been engaged for plasma operation. The capability for continuous operation has been demonstrated from the first experiment runs on by performing 10 ECRH pulses (a few 100 ms) over a single experiment run of several minutes duration (figure 8).

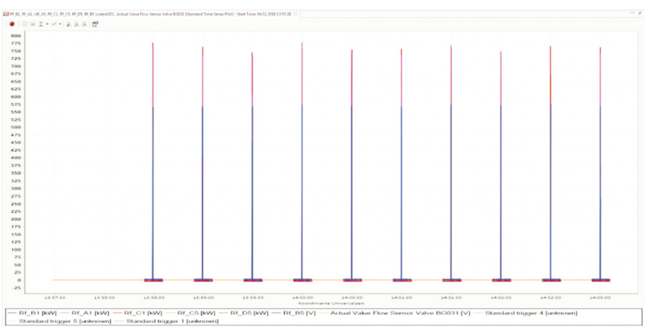

Figure 8: Sequence of 10 ECRH cleaning discharges within a single experiment run.

The system is based on a distributed set of real time computers, which steers many components with a the cycle time of 1 ms. This set contained the ECRH, gas inlet, magnetics, ECE, interferometry, Langmuir probes, the survey spectrometer HEXOS and a subset of the video diagnostic. Control related electronics developments have been performed for the TTEv2 main board (see below) and the piezo valve controller for the gas inlet. The data acquisition system has been operated from the beginning of the commissioning process (April 2014) by collecting continuously data of the local control systems of the vacuum system as well as the cryo and machine instrumentation. The data have been continuously streamed to the archive database for duration longer than two years without any break. While these data have low data rate, the same system archived the high performance data during the experiment runs. Furthermore, this archive has been equipped with a web based interface (WebAPI) for support of the diagnostic systems, which have not been directly implemented within CoDaC data acquisition system. With the help of this WebAPI, local as well as MDSplus based data acquisition systems could be employed for diagnostics while gathering all the experiment data within central archive. All the control and data acquisition systems have been synchronized in time with TTE-System (timer, trigger, event), which consists of a central timer system with its absolute time being adjusted to the GPS time and second version of distributed local timers attached with fibre communication lines (lTTEv2). The standard devices have been equipped with firmware of two different kinds, one for the integration into the segment control system and the other for the provisioning of the seven standard triggers. In this way, the existing hardware could be reused for the non-integrated diagnostics as well. The experiment data archive received in total 10 Gbytes of data per plasma second and approximately 20 TByte in total.

3.3.3 Software Development

The software development has focused on the hardening of the software tools for operation. Beside the development of the aforementioned WebAPI for the web access to all experimental data and the data upload of local and MDSplus data files (7 TByte in OP1.1), the experiment program editor and the configuration database behind have been much improved. This is basically due to the separation of productive and preparation databases, which allow to prepare and to test new component configurations or even enhancements in parallel to Wendelstein 7-X experiment operation. Changes are being made productive in experiment pauses by a strict release procedure, which transfers the new settings into the productive environment and by adapting the existing experiment programs to the new configuration. With these improvements and the improvement of the user interface, the experiment program editor was successfully employed for the creation and execution of approximately 1000 experiment runs in the operation phase OP 1.1. The service oriented architecture for the provisioning of scientific data analyses and modelling functions has been enriched by the VMEC service. This service allows the invocation of MHD equilibrium calculations as well as the access to pre-calculated and validated MHD equilibria. It complements a set of services for the calculation of vacuum as well as equilibrium magnetic fields with accompanying field line tracing and field line diffusion.

3.3.4 Electronics Development

The electronics development has focused on the support for the gas inlet piezo valve driver, development of ECRH signal converters for the real time control, Mirnov amplifiers, TTEv2 base board and the ATCA based data acquisition cards (developed by IPFN, Portugal) including integrators for the magnetic diagnostics. The ATCA cards excellent signal qualities for the ECE channels and the signals of the diamagnetic loops and Rogowski coils.

3.3.5 IT (Information Technology)

The concept of an almost completely virtualized control room worked successfully for the first experiment operation campaign. Scientists and engineers could group themselves at adjacent workplaces while using their personal diagnostics computers. Except for the central control and infrastructure systems, virtual desktops based on VMware's Horizon View have been used throughout. The video wall of the control room is based on 20 times 55 inch LCD screens attached to display nodes, which are solely coupled via 10 Gbit network links to a central video processor (NEC Hiperwall). With this setup, screens from all virtual desktops (but also hardware desktops) could be shared and arranged freely on this wall. Typically during an experiment day, the video wall displayed status diagrams of the vacuum, cryostat, magnets, real time ECRH and gas data, sequence control, session information as well as plasma video data being delivered from the according virtual desktops (figure 9). Although the demand on the number of virtual desktops increased by far more than specified for the high availability storage system, the system worked absolutely reliably with acceptable reaction times.

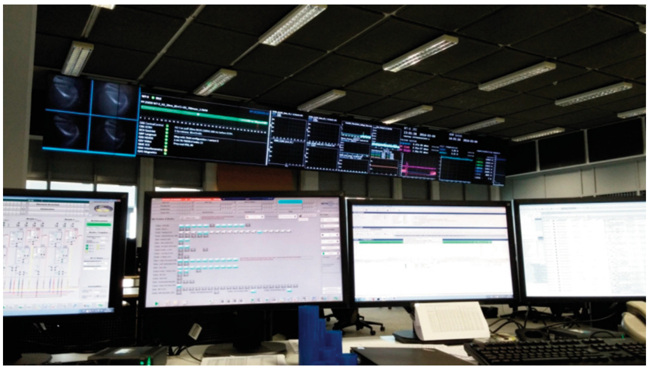

Figure 9: Virtual desktops with control panels (front) and the video wall (back) in the control room.

4 Stellarator Heating and Optimizing (E3)

4.1 Wendelstein 7-X Heating Systems

4.1.1 Electron Cyclotron Resonance Heating (ECRH)

In 2015 seven gyrotrons were installed at Wendelstein 7-X. Due to limitations of the power supplies only six gyrotrons could be operated for OP 1.1. Since Wendelstein 7-X was only weakly armored with first wall components, ECRH operation was restricted to an energy input of 2 MJ, a value which was increased later to 4 MJ. With these boundary conditions the ECRH operation was optimized more with respect to reliability than maximum heating power. The gyrotrons were operated at typically at 0.7 MW each with a maximum total power of up to 4.3 MW. The ECRH system was fully integrated into the Wendelstein 7-X control system and has shown a high reliability already from day one onwards. It routinely provided plasma start-up, core heating and wall conditioning by trains of ECRH conditioning discharges with shot to shot intervals of less than two minutes. The ECRH protective diagnostics were also in operation from day one and have guaranteed efficient and safe ECRH plasma operation. The ECRH operation in OP 1.1 should also be considered as the successful conclusion of the project microwaves (PMW), which ran for more than 15 years and developed and built the ECRH-system as a joint venture of KIT, IPP and IGVP (University Stuttgart).

Status of W7-X Gyrotrons

In 2016 two further gyrotrons were delivered. The TH1507 SN5i gyrotron built by the company Thales (Paris) successfully passed the final acceptance test in June 2016. The one gyrotron built by the company CPI (Palo Alto) was transferred to IPP after showing acceptable performance at the factory test. The site acceptance test has been scheduled for September 2016. The CPI gyrotron is equipped with a new cryo-free magnet, which was successfully tested. The delivery of the final (th tenth) gyrotron, the Thales TH1507 SN8, is scheduled for December 2016 and will complete the ECRH installation for OP 1.2.

Transmission Line, Protective Diagnostic and Control System

The installation of the ECRH-transmission line for the ten gyrotrons is completed. All six operating gyrotrons have been aligned up to the plasma vessel, where the beam footprints on the heat shield were measured with an infrared-camera. The measured transmission efficiency is ~ 94%. For these measurements a retro-reflector mirror was placed at the end of the multi-beam transmission line that sent the beam back into the same load, which had been used to measure the power before passing the multi beam section. The resulting transmission losses were close to the theoretical values. After OP 1.1, two remote steering launchers were installed into the vessel as shown in figure 10.

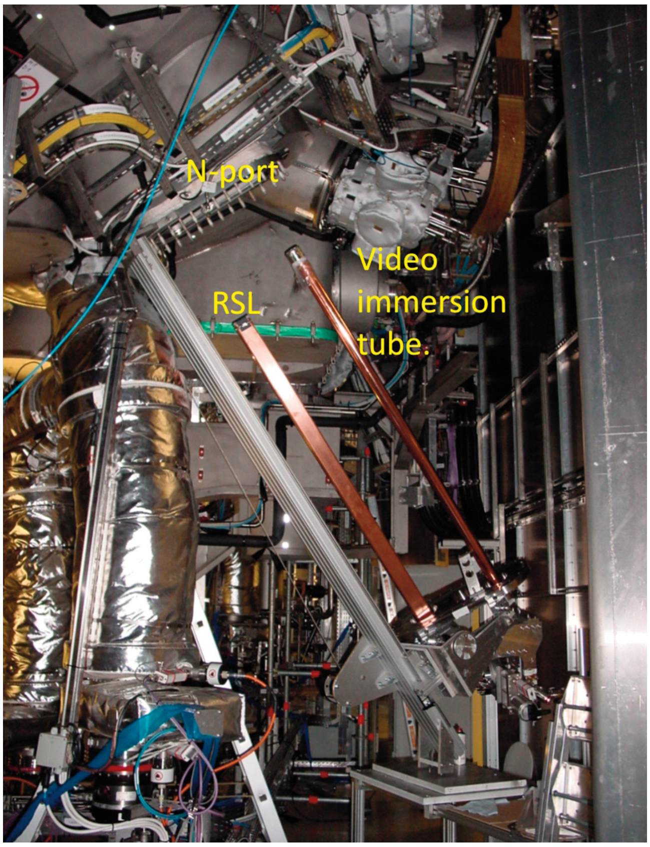

Figure 10: Remote steering launcher (RSL) before being inserted into the 1-AEN10 port.

Design and manufacture of these complex components were funded by the federal ministry (Bundesministerium für Bildung und Forschung, BMBF) as a separate project. They enable more sophisticated heating and current drive scenarios in poloidal planes with almost zero-magnetic field gradient. Wendelstein 7-X is used as a full performance (1 MW, cw) test-bed for this new DEMO-relevant launcher concept.

Operation of the six gyrotrons was fully controlled by the central control system (cOPM). The experimental requirements like timing, power and modulation were automatically transferred into the gyrotron parameters. All relevant operation data are stored into the Wendelstein 7-X archive. Safe and reliable operation is guaranteed by the ECRH protective diagnostics. The most important one is the microwave stray radiation measurement in the plasma vessel. These so called sniffer probes have been absolutely calibrated with the stray radiation in the empty plasma vessel. They gave a very reliable signal of the ECRH absorption and were used as the one and only plasma interlock for first Wendelstein 7-X operation. A characteristic signal is shown in figure 11. The stray radiation decreases during plasma start-up exponentially within less than 15 ms due to the build-up of a well absorbing core plasma. The discharge used as an example was terminated by a sniffer interlock indicating upcoming microwave stray-radiation due to a decay of the plasma which in this particular case was caused by a radiative collapse. The decrease of the core temperature reduces the ECRH absorption and thus the stray radiation level is increasing. When the interlock threshold is reached, the ECRH is terminated in order to avoid damages by non-absorbed power. The electron-cyclotron absorption (ECA) diagnostic, which measures transmitted power in the position opposite to the ECRH launcher, was used to control the ECRH beam direction and polarization. It also provided a direct measure of the ECRH absorption. As a further protective diagnostic each launcher is equipped with a video camera in an immersion tube, which detects arcing and overheating of heat shield tiles and observed the limiter in OP 1.1 as well. Finally, three selected port bellows were equipped with thermo elements and ECRH-bolometers to measure their temperature increase due to ECRH stray radiation.

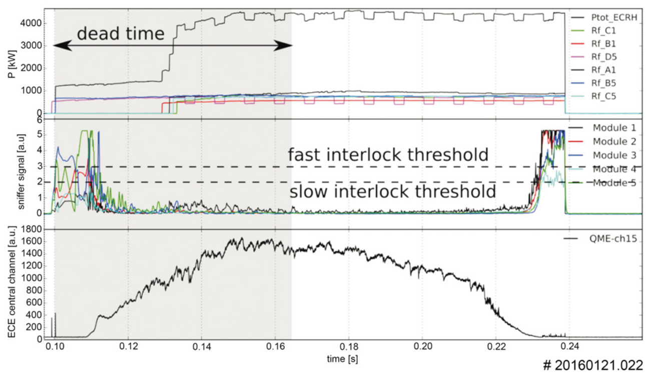

Figure 11: From the top: ECRH power, ECRH-stray radiation (sniffer), and central electron temperature. At the plasma start-up at 0.1 s the ECRH is only weakly absorbed and the stray radiation is high until a high temperature core has been build up. Therefore the fast sniffer interlock had been deactivated (dead time) for first 55 ms.

ECRH- and ECCD Physics Experiments at OP1.1

The first Wendelstein 7-X operation phase OP 1.1 was considered as a commissioning phase for all Wendelstein 7-X systems. As the ECRH system itself had already been intensively commissioned in advance, it was available with its full variability from the first day and many ECRH physics experiments could be performed. The first main issue was plasma generation. Even though the vacuum and impurity conditions of Wendelstein 7-X were rather poor, plasma start-up could be initiated with typically 1MW ECRH power using the second harmonic extraordinary mode (X2). The sniffer signal turned out to be the most reliable and meaningful diagnostic to indicate plasma generation. When the signal dropped below 10% of its maximum level a well absorbing hot plasma core existed. This was achieved typically

8-15 ms after the gyrotron beam was launched. The hot core then expanded up to the limiter and the density could be increased. In the first operation days the hot plasma phase was terminated by a radiation collapse, where strongly enhanced impurity radiation reduced the core temperature. Once the core temperature dropped below a level where the 140 GHz beam is well absorbed (T_e<0.4 keV) the sniffer signals exceed the defined plasma interlock threshold and the ECRH was terminated (see figure 12). The wall conditions and this plasma duration could be increased by a combination of repetitive ECRH conditioning discharges and helium glow discharge cleaning in-between the operation days (when the magnetic field was turned off). Central high power ECRH with up to 4.3 MW enabled to achieve electron temperatures above 8 kV (for physics details see the paragraph on OP1.1 objectives and results below). These high electron temperatures at moderate densities of $1\text{-}4 \cdot 10^{19}$ m^{-3} enabled to test the ECRH operation at the second harmonic ordinary mode (O2), which is foreseen for high density operation above the X2 cutoff density of $1.2 \cdot 10^{20}$ m^{-3} in the next operational phases of Wendelstein 7-X, where good confinement and divertor performance are expected. The single pass absorption for an O2-polarized beam, measured with the ECA-diagnostic, was 70%. With multi-reflections the total O2 absorption was estimated from the sniffer signals to be over 90%, and a completely O2-sustained plasma could be demonstrated (figure 12).

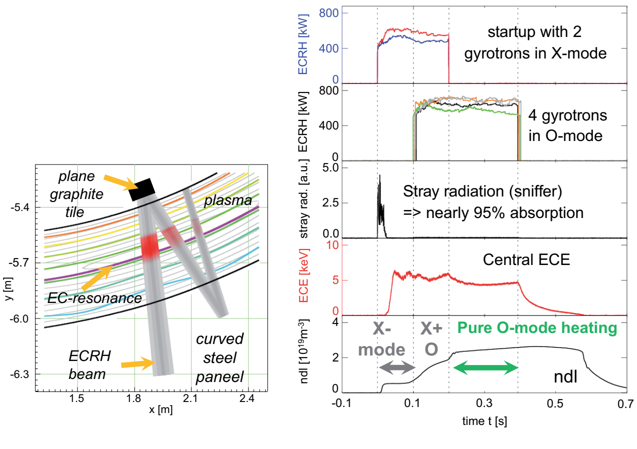

Figure 12: Demonstration of solely O2-heated plasma. From the top ECRH power, Stray radiation and power absorption, electron density and electron temperature. The plasma was created by X2-ECRH and O2-ECRH takes over after 0.2 s.

First ECCD experiments could be performed as well. However, stationary current profiles could not be reached as the achievable pulse length of about 1s is much smaller than the characteristic L/R time (30-100 s) for the toroidal plasma current evolution. The ECCD current therefore was nearly completely screened by the induced counter currents. However, the change of the internal iota profile was high enough even to arrive at resonant values like iota=1. These estimates go in line with observations of large crashes (up to a few 10% of initial values) of the centrally peaked electron temperature. The crash frequency could be controlled by the ECCD power. In contrast to tokamaks, there is a strong central heating by ECRH, which recovers the electron temperature on a fast time scale (10 ms) while the current distribution recovery time is of the order of 100 ms, which determines the crash frequency.

4.1.2 Neutral Beam Injection (NBI)

The neutral beam injection (NBI) system is planned to go into operation during operational phase OP 1.2a. The system consists of the two injector boxes NI20 and NI21 and the required peripheral components and media interfaces. Both injector boxes are installed in the torus hall symmetrically to the triangular shaped plasma in module 2. They can be equipped with up to four positive ion neutral injectors (PINI) each, however, in OP 1.2a und 1.2b only two PINIs will be installed on each injector. In OP 1.2 it is planned to operate the NBI system with hydrogen only, even though deuterium operation also would be technically possible. The high voltage grid spacing of the acceleration system has been chosen to produce hydrogen beams at 55 keV or deuterium beams at 60 keV, this corresponds to injection powers of 1.7 MW and 2.5 MW per PINI, respectively. It is planned to commission both injectors before and during OP1.2a, so that at least NI21 is available for plasma experiments towards the end of OP1.2a. The design of the injector boxes with all its internal components and the design of the PINIs are almost identical copies of the NBI system on ASDEX Upgrade. These components have been designed, procured and built by the NBI team in Garching and delivered to the Greifswald site. All additionally needed peripheral components are designed and procured mostly at the Greifswald site. As of summer 2016 both injector boxes have been placed and aligned in the torus hall and connected via bellows and gate valves to the assigned ports in module 2. These injector boxes house the neutralizer, the ion beam dump, the deflection magnet to guide the accelerated hydrogen ions that have not been neutralized onto the ion beam dump, the vertically movable calorimeter, various diagnostics and titanium sublimation pumps as a fast pumping system to prevent neutralizer gas to stream into the plasma vessel. Differential pumping between the neutralizer region and the duct region is achieved by gas shields inside the injector boxes. Each titanium sublimation pump consists of nine, about 4 m long three-sided chambers of water-cooled, corrugated aluminium panels. In each chamber three pairs of titanium wires are hanging which are heated sequentially (figure 13). Three of the four required titanium pumps have been assembled and leak tested on a custom made assembly scaffold, the installation of the titanium wires is in progress.

Figure 13: Assembly of the titanium sublimation pumps in the storage scaffold.

A dedicated scaffold was built for maintenance and storage. To initiate the titanium sublimation pumps a sufficiently high current is run through the wires so that some titanium is evaporated and de-sublimated on the corrugated surface of the panels. This conditioning has to be repeated whenever the pumping capability is no longer sufficient. During plasma operation this has to be done after each plasma experiment with NBI. The NBI team in Garching has developed pump conditioning with alternating current (AC) in an ambient magnet field. Based on their results AC power supplies have been specified, procured and ordered. Most of these components require water cooling and have electrical feed throughs that are prone to leaks. Both, the calorimeter and the deflection magnet initially did not meet the required leak rates. These problems could be solved by taking both components apart and diligently re-assembly with several leak checks in-between to better identify critical areas. The neutral beam system requires a number of peripheral components that predominately are located outside of the torus hall. The plasma sources for all PINIs are powered by inductively coupled high frequency currents with 120 kW at 1 MHz. For the plasma sources of NI21 two radio frequency (RF) solid-state amplifiers were specified, procured and ordered. After successful company acceptance test they were installed on site, the required cables were ordered. The on-site acceptance test will take place after they have been electrically connected to the plasma sources. In summer 2016 it was decided to operate the plasma sources of NI20 with identical RF generators rather than with the existing free oscillators that were in operation at the radial injector at Wendelstein 7-AS in the past. These RF generators were ordered as well. The power supplies for the magnets have been specified and ordered and are expected to arrive in the fall of 2016. The support unit of the calorimeter includes a commercial crane to move the calorimeter up and down while under vacuum in the NI injector box. Legal permission has been obtained to operate the crane; some further documentation is required to operate the calorimeter crane when mounted on the NI injector box. The grid system of each PINI is separately powered with high voltage that is provided by the IPP high voltage system. The high voltage lines can be grounded in a NBI high voltage box inside the torus hall. The details of the switches and the required control system have been specified and some of the components have been installed. Each injector box is equipped with a vacuum system consisting of turbo molecular pumps at the boxes, roots pumps and roughing pumps in a separate pump stand. The pump stand is equipped with a set of spare roots and roughing pumps. The design of the vacuum lines from the vacuum pump stand to the turbo pumps has been completed and the needed parts have been ordered. The design of the pressurized dry air distribution system has been completed and the distribution lines, valves and sensors have been installed. The design of the gas distribution system including the line evacuating system for gas change has been completed and installed. The operation of the NBI system will be done via "Siematic" control using PCS7 supplemented with a "Pilz" component safety system. Both injectors of the NBI system have separate, but identical control systems and individual interfaces to the central operations control and the central safety system. The modes of operation of each NBI system correspond to those of the NBI at AUG. The control of each subcomponent is done via individual program modules in a hierarchical manner. The functional requirements of the program modules are defined for most subcomponents. The programming is in progress and partially supported by the Garching team. The component safety requirements were derived from the NBI safety analysis. The design of the electronic circuitry has been completed.

The commissioning of the NBI system will be done in several steps. According to this schedule NI21 conditioning will start in summer 2017; NI20 conditioning will commence somewhat later, such that towards the end of OP 1.2a the first pulse with NBI in the plasma can be performed.

4.1.3 Ion Cyclotron Resonance Heating (ICRH)

The ion cyclotron resonance frequency heating (ICRH) system is planned to go into operation in operational phase OP 1.2b. The system consists of one high frequency (HF) generator, coaxial transmission lines with a matching network and a movable antenna. The antenna is mounted in the torus hall in module 2 near the plane where the Wendelstein 7-X plasma has a bean-shaped cross-section and a nearly tokamak-like magnetic field profile. The antenna is powered via a transmission line; all other components are set up the ICRH hall. It is expected that up to 1.6 MW of HF power can be coupled to the antenna for up to 10 seconds. The system is designed to operate at frequencies between 25 MHz and 38 MHz to facilitate the following heating mechanisms at a nominal magnetic field of 2.5 T on axis: hydrogen minority, helium 3 minority, three-ion heating of a hydrogen/helium mixture with a minority of helium 3. The main purpose of the ICRH system is to provide high energy test particles that can mimic the behavior of fusion alpha particles in a fusion reactor. The system is designed, procured and fabricated within a collaboration between IPP, ERM/KMS Brussels and FZ Jülich. The HF generator and the transmission line matching system are components that were in use at Textor. The HF generator was moved to IPP Greifswald. Work started to upgrade its electronics and control system to the new requirements derived from changed regulations and interface to the Wendelstein 7-X control system. The routing of the transmission line between the ICRH hall and the antenna was developed. The trail assembly of a section of the transmission line was successfully completed by the assembly department. The remaining transmission line parts and supports are being ordered. The antenna was designed specifically for Wendelstein 7-X by ERM/KMS. Its main features are: Two independently fed current straps with in-house prematching via variable capacitors, radially movable by about 30 cm, plug-in type design, all parts actively water-cooled. In summer 2016 the conceptual and detail design was nearly finished. The fabrication process is challenging due to the three-dimensional shape of the components that require embedded water-cooling channels which requires three-dimensional milling machines and electron beam welding. Fabrication steps were qualified wherever needed. The fabrication started. After assembly the antenna will first be mounted into a test stand at FZ Jülich for vigorous testing of functionalities of the antenna including high voltage standoff, moving, baking, water cooling, vacuum leak tightness. The ICRH system requires a number of peripheral components to provide the supply with cooling water, high voltage, data acquisition and control. These subsystems are in their conceptual design phase.

4.2 Wendelstein 7-X Core Diagnostics

For the first experiments three core diagnostics were considered to be mandatory, the neutron counters required for the operation permission, the 32-channel ECE radiometer to characterize the ECRH heating and a first density measurement from interferometry. Together with a ten-channel setup of the Thomson scattering system and the X-ray imaging spectrometer these core diagnostics provided the profiles for T_e, n_e and T_i as shown in figure 14. A first measurement of the core radial electric field from X-ray impurity lines was also possible.

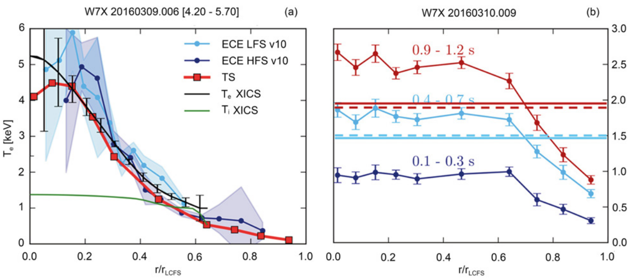

Figure 14: Representative temperature and density profiles: Left: electron- and ion-temperature measured by ECE, Thomson scattering and X-ray imaging of Ar impurity lines (XICS). Error bars of ECE at low- and high-field side spectra resulting from absolute calibration uncertainties are shaded. Thomson data plotted without error bars as systematic errors are being investigated; statistical errors are negligible. XICS error bars (black lines) represent the inversion uncertainty. Right: Electron density measured by Thomson scattering during the build up phase of the plasma (dark blue) and two subsequent density steps. The resulting average densities are plotted as horizontal bars (continuous line) and compared with the value derived from interferometry (dashed line). The limiter position (LCFS) corresponds to r_{eff}=50 cm.

Neutron Counters

Operation of calibrated neutron counters was an indispensible requirement to obtain the operation permission of Wendelstein 7-X. Three probing locations around the torus have been selected with four detectors each to reach a nearly energy-independent response for the entire energy range from thermal neutrons up to at least 2.5 MeV in view of later deuterium operation. The monitors are designed for the expected neutron yields in deuterium plasmas from 1011 to 1016 neutrons per second with a time resolution of 5 ms and a statistical uncertainty of better than 15%. Prior to OP1.1 the system has been absolutely calibrated in cooperation with the German "Physikalisch-Technische Bundesanstalt" (PTB) by moving a radiation source of known strength in the vessel toroidally along a railway system that roughly follows the later plasma core.

These measurements were supported by Monte-Carlo calculations (MCNP) of the neutron propagation using a simplified model of the mechanical structure of Wendelstein 7-X. For the first experiments the system was prone to sporadic phases of virtual counts on a very low level, independent of Wendelstein 7-X operation, resulting from electromagnetic perturbations.

Electron Cyclotron Emission Diagnostic (ECE)

The Electron Cyclotron Emission diagnostic, ECE, was operated throughout OP 1.1 as the main tool to study electron heating by ECRH and the subsequent electron heat transport. It measures the 2nd harmonic X-mode emission (X2) in the frequency band from 126 GHz to 162 GHz with a 32-channel absolutely calibrated heterodyne radiometer. Calibration is performed by a second, identical Gaussian optical system as a twin outside the torus, however with a hot-cold calibration source chopping between LN_2 temperature and room temperature. The overall absolute calibration error is estimated to be ~10% with outliers for individual channels due to low detection sensitivity. This radiometer is supplemented by an additional 16 channel "zoom system" in parallel to the standard filterbank that measures a 4 GHz wide frequency range with higher resolution. Besides the blackbody emission from optically thick plasma layers from which local radiation temperatures can be derived, the ECE spectra show additional spectral features e.g. resulting from higher energetic thermal electrons for which the plasma optically grey. Next to stationary T_e-profiles a variety of dynamic phenomena was studied such as ECRH power switching and power modulation, providing heatwaves, mode activity, temperature crashes and fluctuations.

Interferometry

For later long-pulse operation a Dispersion Interferometer has been developed that uses a CO_2-laser, and frequency doubling crystals. The measured phase shift to first order is independent on vibrations and drifts along the laser beam path, even an intermittent signal loss can be tolerated. The beam path is nearly identical to those of the Thomson scattering which eases cross calibration and allows for a retroreflector outside the vessel. The continuous real time phase measurement is achieved by a phase modulation scheme and direct digital sampling followed by a FPGA-based data evaluation. The system was successfully commissioned during OP1.1 and delivered line integrated densities for most discharges. All diagnostic components are located in the torus hall, thus being prepared for continuous and fully remote operation, which becomes necessary when in OP1.2 the diagnostic will not be accessible for several days. In preparation of a later multichannel system, laboratory and in-situ tests of the mandatory high-heatload retroreflectors for this diagnostic are being prepared for OP 1.2.

Thomson Scattering

The Thomson scattering system which uses a Nd:YAG laser with a repetition rate of 10 Hz started operation with a reduced set of 10 spatial points (radial resolution 2.5 to 4 cm), covering a half profile with one data point located already in the limiter shadow. Laser, polychromators and the data acquisition system are located in diagnostic rooms outside the torus hall and the beam is guided by mirrors to the entrance port at Wendelstein 7-X and absorbed behind the exit port in a beam dump. Two observation optics image the scattered light onto fiber bundles. High dynamic range ADCs with 14 bit resolution together with 1 GS/s sampling rate are used to digitize the signals from Si-avalanche diodes. For the density calibration rotational Raman scattering off Nitrogen gas at various pressure is used. In most cases the derived average densities agree with the interferometer within 10%.

X-ray Imaging Crystal Spectrometer and High Resolution X-ray Spectrometer

Two X-ray imaging spectrometer systems, the X-ray Imaging Crystal Spectrometer (XICS) and the High Resolution X-ray Imaging Spectrometer (HR-XIS), have been commissioned in cooperation with the Princeton Plasma Physics Laboratory and the Forschungszentrum Jülich respectively. Both systems are designed to provide radial profiles of T_i and T_e, as well as densities and plasma rotation velocities for selected impurities from which the core radial electric field could be estimated. The XICS system using Ar as a tracer provided data for many discharges being the only Ti measurements: The High Resolution X-ray Imaging Spectrometer (HR-XIS), was tested successfully only at the end of the campaign.

Diagnostic Preparation for OP1.2

For the physics studies in OP1.2 improved profile diagnostics are being prepared. The Thomson scattering diagnostic will be upgraded to 80 channels (until the end of OP1.2) covering the full profile and an additional laser to achieve a 50 ms time resolution. For continuous and fast edge density profile measurements Thomson scattering will be supplemented by an ultrafast frequency modulated profile reflectometer located in close neighborhood to the Doppler reflectometer and the Li-beam edge density profile measurement. For further T_i and E_r information first charge exchange spectroscopy (CXRS) measurements are being prepared using one of the NBI beams which will be commissioned during OP1.2. Ultimately, two observation ports are being used to provide sufficient spatial resolution and accuracy for poloidal and toroidal flow measurements. The active system will be supplemented by passive CXRS using impurity lines.

4.3 Wendelstein 7-X Operational Phase 1.1 Objectives and Results

4.3.1 Goals and Achievements

First plasma operation of Wendelstein 7-X has been successfully conducted on December 10th, 2015 once the operation permit was granted. Hydrogen plasmas were created from February 5th, 2016 onwards. The main goals of the first experimental campaign of Wendelstein 7-X were the demonstration of good flux surfaces, the integral commissioning of the device including the cryo-plant, the magnet system, control and data acquisition, heating and diagnostics. The demonstration of these goals was expected to culminate in first plasma operation with more than 10 discharges at electron temperatures >1 keV at low densities ($5 \cdot 10^{18} m^{-3}$). These goals have been clearly achieved in the first days of operation. Successive commissioning of more than 20 diagnostics systems and progress in wall conditioning allowed to successively and safely increasing the plasma performance. The campaign ended on March 10th, 2016 and more than 2000 plasma discharges have been performed and a comprehensive scientific program with more than four hundred scientific discharge programs could be performed.

4.3.2 Plasma Generation

Prerequisite for plasma operation was reliable plasma start-up. The available heating power at 140 GHz was in total 4.3 MW for X2- and O2-mode operation (resonance at 2.5 T). Despite initially very poor vacuum and impurity conditions, plasma start-up could be realized with about 1 MW ECR heating from day one as described above. Stray-radiation diagnostics were successfully employed to provide basic ECRH interlock systems for safe operation. These signals also allowed the optimization of plasma start-up in terms of heating power, gas pre-fill and optimization of required timing.

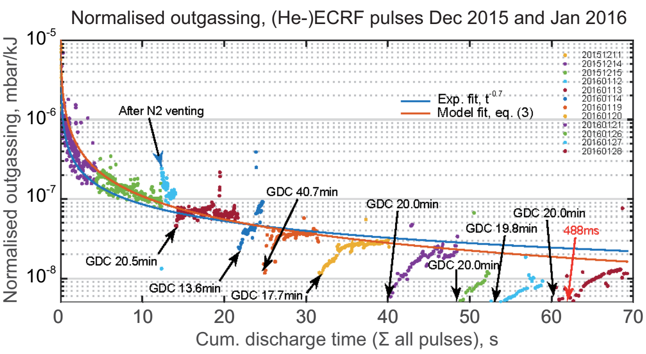

Figure 15: Normalised outgassing during He-operation: experimental data (dots), and fits to a model discussed in Wauters et al., EPS 2016, Leuven. The discontinuities in the outgassing trend are result of He-glow discharges (GDC) with magnetic field turned off.

First plasmas in helium were limited to about 50 ms discharge length at central T_e ~1 keV and T_i < 1 keV at average densities of $2 \cdot 10^{19} m^{-3}$. Basic impurity content monitoring indicated unfavorable wall conditions inducing radiative collapses. Wall conditioning techniques have been applied which allowed a continuous prolongation of the plasma pulse lengths up to 500 ms (T_e ~ 8 keV and T_i ~2 keV). Progress in the wall conditioning is reflected in figure 15. The largest decrease of outgassing were the result of He glow-discharge conditioning and ECR conditioning discharge sequences with helium. The latter was employed with the magnetic field applied. With these techniques, long and stationary discharges became accessible, but outgassing turned out to be a major impurity source continuously increasing over the plasma operation days (main observed impurities were O,C, S, Cl with indications for Fe and Cu).

4.3.3 Core Confinement, Transport Studies

Given the comprehensive set of diagnostics for measuring plasma profiles, even in the first weeks of operation first confinement and transport studies could be conducted. Figure 14 shows electron and ion temperature profiles with fairly good agreement between Thomson, ECE and X-ray imaging spectroscopy data. An important finding, consistent with global diamagnetic energy measurements, are energy confinement times of τ_E~100 to 150 ms. The measured values are consistent with the International Stellarator Scaling ISS04 with highest observed configuration factors, but are still below neoclassical predictions. The plasma profiles in figure 14 show centrally peaked high electron temperatures in on-axis ECR heated plasmas much higher than ion temperatures. High power ECRH with up to 4.3 MW enabled to achieve electron temperature up to 8 keV. The sustainment of these high, but not equilibrated electron temperatures requires a substantial reduction of electron fluxes. In the stellarator specific, 1/ν transport regime, the expected energy fluxes would exceed the applied heating power by almost an order of magnitude. Instead, a positive radial electric field leads to a reduction of the transport coefficients which is consistent with the observed profiles (figure 16). Altogether these are characteristics of the so-called central electron root confinement regime (CERC) expected for electron heated stellarator plasmas. CERC characteristics at low density ECRH discharges – i.e. peaked T_e and flat T_i profiles together with a positive E_r allow the first comparison of transport and confinement in Wendelstein 7-X with its neoclassical predictions. As the peaked T_e-profiles provide pronounced pressure gradients deep in the plasma core, they also will be the first candidate to measure the Shafranov shift, which as a further optimization criterion of Wendelstein 7-X should be reduced. With off-axis heating the Te profiles broaden considerably as expected for a local transport model with no indication for profile stiffness. Heatwave propagation experiments, with on- and off-axis power modulation of the gyrotrons, allow for the determination of the ECRH power deposition profiles as well as for dynamic electron transport studies.

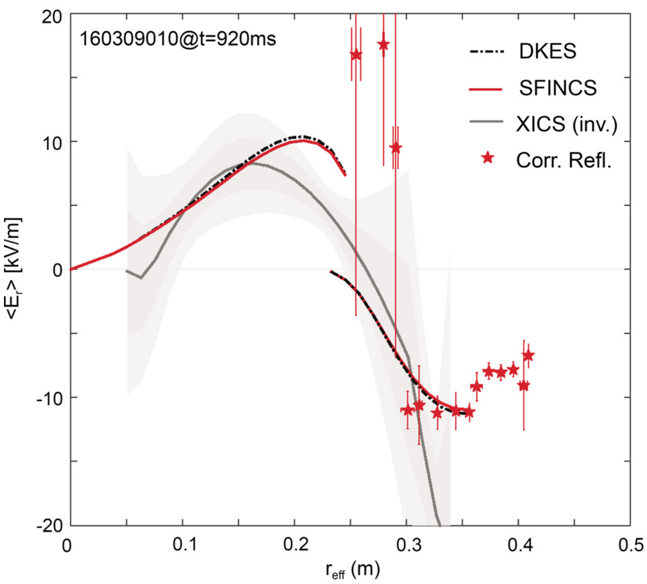

Figure 16: The figure compares the expectations from neoclassical theory on how large the required radial electric field needs to be (DKES, SFINCS) and measurements of the poloidal rotation and the resulting mean radial electric field (reflectometry, XICS). It is found that the central confinement region shows so-called Central-Electron-Root-Confinement (CERC) which appears to be the characteristic plasma scenario for most of the discharges in the first experiment campaign. In a first limited scan of the toroidal magnetic mirror, indications for changes in the transport in response to the magnetic field according to neoclassical expectations have been found.

Moreover, these first experiments already displayed a rich phenomenology of dynamic phenomena such as a nearly coherent activity with frequency around 7 to 10 kHz located ~r/a=0.5 in several scenarios and individual transport events which are candidates to affect confinement. An example is given in figure 17.

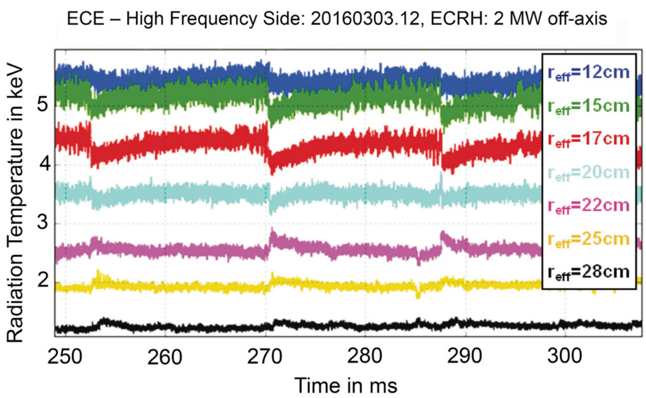

Figure 17: Time traces of ECE channels probing the electron temperature at various radii at the high field side display sudden crashes of the electron temperature with inversion radius at about r_{eff}=20 cm i.e. r/a~0.4. The discharge is off axis heated by 2MW ECRH.

5 Stellarator Edge and Divertor (E4)

During the first operation phase OP 1.1, the plasma-wall interaction was realized with five uncooled, inboard graphite limiters. A magnetic configuration was selected with ι_o =0.79 at the magnetic axis (plasma center) and ι_a = 0.87 at the plasma edge, so the the 5/6≈0.83 island chain was shifted clearly into the plasma, and the 5/5 island chain shifted far outside the plasma. This provides closed flux surfaces closely inside the limiters and beyond, such that diverting island structures at the plasma edge are avoided. Only a small configuration variation was acceptable to minimize the risk of uncontrolled loads apart from the limiters. To avoid overheating, the maximum allowed absorbed energy was set to 4 MJ per experiment program – thus restricting the discharge length. Stationary discharge scenarios over 6 s were achieved, however with P_{ECRH} lowered to 0.6 MW over most of the pulse duration due to this energy limit. For plasma breakdown and heating a total power of 4.3 MW ECRH was routinely provided by six long-pulse 140 GHz gyrotrons.

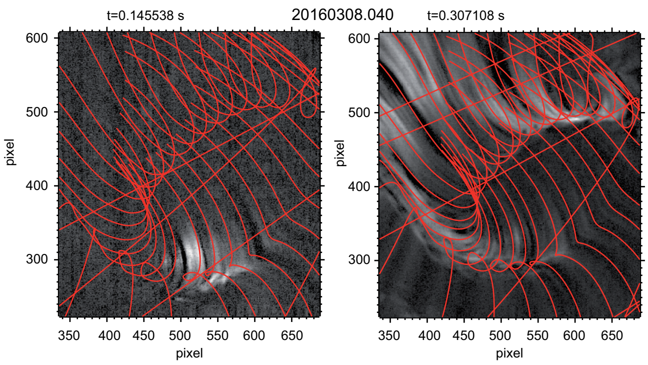

Figure 18. Filamentary structures observed at the plasma edge during strong gas puffing.

The typical evolution of a discharge observed by the video cameras (see figure 18) starts with a small radiating magnetic surface around the hot plasma core where the ECRH is absorbed. The radiating surface expands until the last closed flux surface defined by the limiters is reached. Depending on the wall conditions this plateau phase may last up to 6 s. The discharge is terminated by either switching off the heating or by impurity accumulation released by the first wall components causing a radiation collapse. Taking images with exposure time shorter than 200 µs, filamentary structures can be observed at the plasma edge where the visible radiation is intense enough for the short exposure detection. Since no filter was applied, we can only speculate about the origin of the enhanced radiation of the filaments. Assuming that the electron impact excitation of neutral or low-ionization-stage ions is the dominant process for creating the light, the enhanced light intensity is very likely due to an increased

electron density inside the filaments. The filaments are most prominent during the radiation collapse and the breakdown both in helium and hydrogen plasmas, but they were also observed in the steady-state phase, in particular when gas puffing was applied.
With the carbon tiles not yet attached to the heat shields and the already installed divertor frame being unprotected, particular care had been taken in designing a limiter magnetic configuration that ensured that these components were not exposed to any convective loads. With the limiters being simply attached to the heat shield structures we expect to be able to run one second discharges at 4 MW of heating power (ECRH), assuming the heat loads can be spread out evenly across the limiters. As already mentioned, the magnetic vacuum configuration had been chosen such that it had a smooth scrape-off layer, with no stochastic region and no large magnetic islands, such that the limiters would intercept more than 99% of the convective plasma heat load in the scrape-off layer (SOL). The typical connection length of magnetic field lines in the SOL was on the order of a few tens of meters and consisted of three separate helical magnetic flux bundles with different connection lengths. Therefore, these separate heat flux channels were expected to give rise to localized peaks in the limiter power deposition patterns. EMC3-Eirene modelling of OP1.1 plasmas shows strong impact of the magnetic topology on pressure and flow profiles.

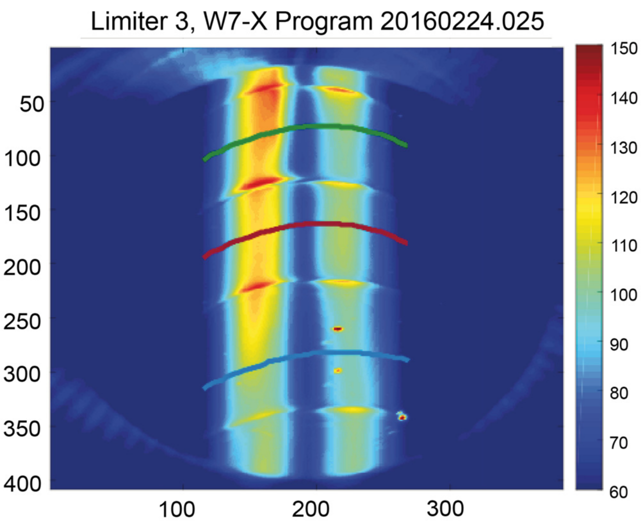

Figure 19: Thermal image of the central part of limiter in module 3 taken with an infrared camera during one discharge. Colour scale indicates surface temperature in Celsius centigrade.

In figure 19 a temperature distribution on part of limiter 3 is presented. Two strike lines on both sides of watershed area are clearly present. The heterogeneous temperature distribution is an effect of 3D topology of the scrape-off layer and the leading edges of the limiter tiles. During the experiments the surface temperature could reach up to 1000 °C at some locations during the high performance scenarios, which corresponds to heat fluxes of up to 5 MW/m^2.

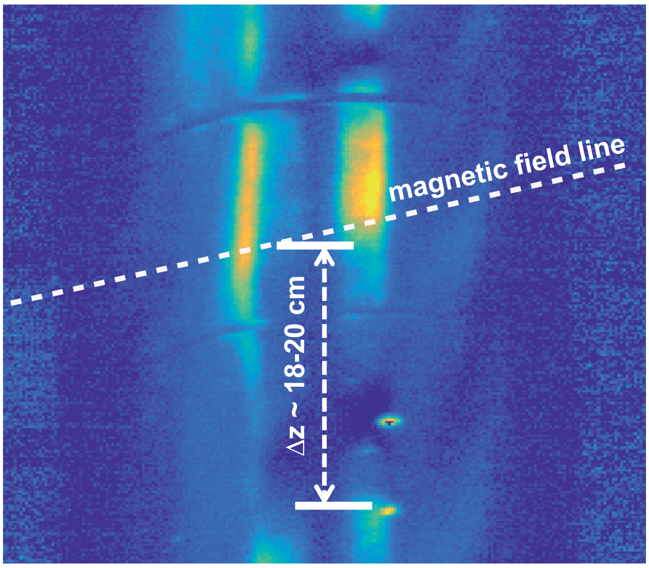

Figure 20: Filamentary structure of transient power loads observed by visualizing a temperature increase between two frames on the surface of limiter in module 3.

Similar to video cameras also fast infrared cameras measured filamentary structures appearing at the plasma edge. Figure 20 shows an example of such structures on the surface of limiter 3. The image shows the temperature difference between two consecutive frames, i.e. an increase in surface temperature due to filamentary transient loads. The typical structure observed during the OP1.1 campaign consists of two or three hotspots on both sides of the watershed, separated by a distance of appr. 18-20 cm, corresponding to poloidal mode number of 15-16. The peak transient power loads can locally exceed the steady state values by a factor of 3. Several diagnostics have been developed in order to characterize the scrape-off layer. In particular, two sets of Langmuir probes were integrated into the limiter of module 5, both – an upper and a lower array – consisting of 20 probe tips each. This allowed for measurement of the detailed distribution of electron temperature and density at the limiter. An example from a hydrogen discharge is presented in figure 21. Comparing the upper left and lower right sets of probes, which were measured within the same flux tube, shows an asymmetry in electron temperature and density. The mechanism is not clear at the moment, but it is expected (based on results from Wendelstein 7-AS and similar observations in tokamaks) that E×B drifts lead to asymmetric distribution of SOL parameters.

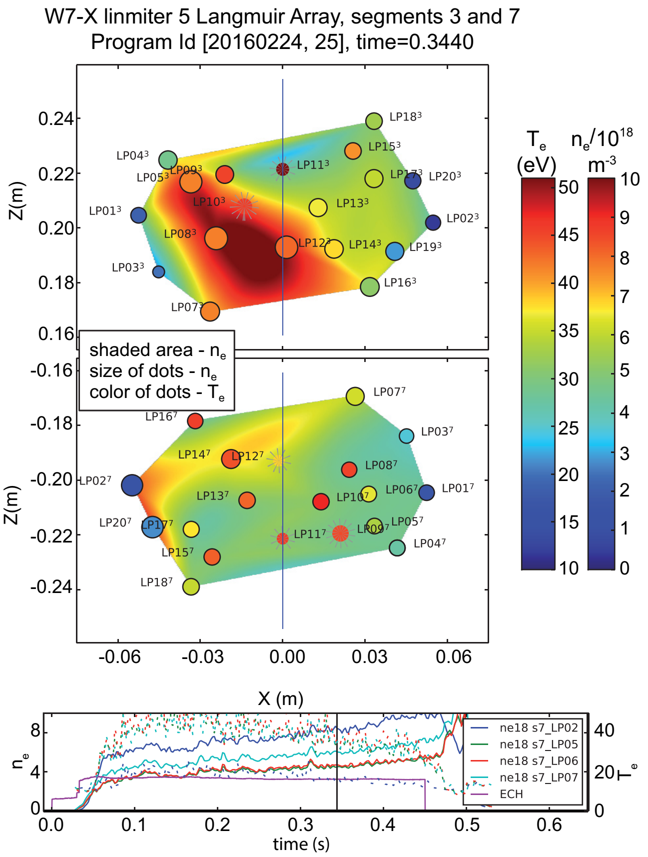

Figure 21: Distribution of electron temperature and density as measured by Langmuir probes embedded in limiter 5. Upper graph shows results from tile 3, lower one from tile 7. The position of the dots indicates the position of the Langmuir probes on a limiter surface. The colour scale indicates electron temperature (dots) and interpolated electron density (shaded area). The size of the dots is proportional to electron density as measured by individual probes.

The radial distribution of electron temperature and density has been measured with fast, reciprocating probe mounted as a part of multi-purpose manipulator in module 4. The probe head includes two 3D magnetic pick-up coil arrays, five Langmuir probe pins and a Mach probe. This allows simultaneously measuring the edge radial profiles of the magnetic fields, the electron temperature and density, the electric field, and the plasma flow. An example of measured electron temperature and density is shown in figure 22. The last closed flux surface for this configuration is located at major radius of R=6.02 m. The estimated SOL width from the decay length of T_e and n_e is on the order of 1 cm.

After the commissioning of the superconducting coils and well before the first plasma, the vacuum magnetic flux surfaces were measured. As mentioned in the preceding section, the OP1.1 configuration had ɨ varying between 0.79 and 0.87, so that two low-order resonant flux surfaces at ɨ=4/5 and 5/6 were present. For the latter, an island chain with 6 poloidal lobes should exist.

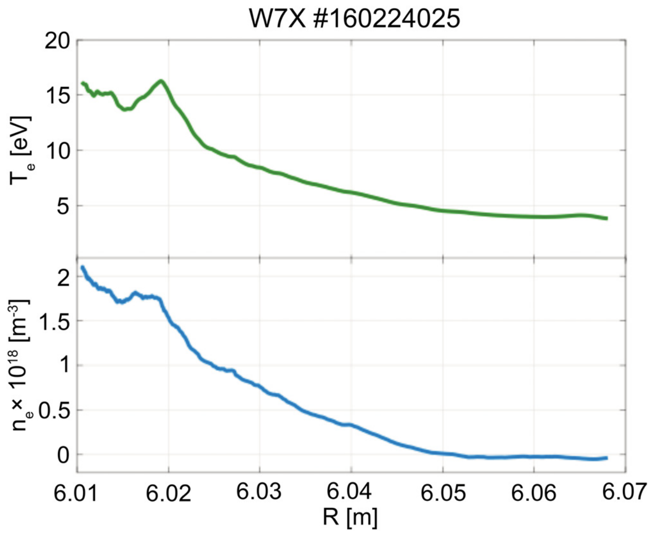

Figure 22: Radial profiles of electron temperature (top) and density (bottom) measured by Langmuir probes mounted on the reciprocating multi-purpose manipulator.

By means of vacuum magnetic flux surface measurements, closed and nested flux surfaces were detected from the magnetic axis up to the last closed flux surface were detected, as was the 5/6 island chain. Figure 23 (left) shows a set of nested flux surfaces generated by sweeping the fluorescent rod from close to the magnetic axis up to the 5/6 island chain. The shape of the flux surfaces as well as the position and symmetry of the island chain were in agreement with expectations from numerical simulations. For the 5/6 island-chain, the field line close to the X-point is visible as well because of a higher background pressure during this measurement. Thus field lines could be detected for about 40 toroidal transits, corresponding to an electron beam length of more than 1 km. The measurements also confirmed the expected elastic deformations of the non-planar field coils arising from the electromagnetic forces in the ambient magnetic field.

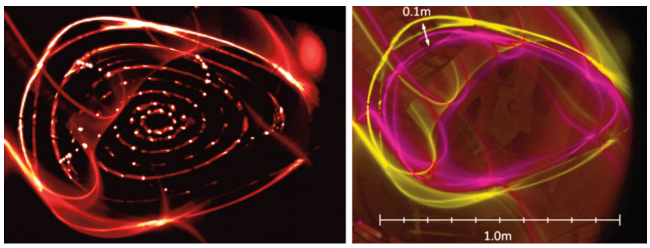

Figure 23: Left: Six selected magnetic flux surfaces of the OP 1.1 limiter configuration at 2.5 T. At the 5/6 island the field line close to the X-point is visible as a results of the background pressure during the measurement. The bright spot on the upper right hand corner is due to reflections of in-vessel components in front of the camera port. Right: Superposition of the 5/6 island chain for the OP 1.1 limiter configuration at 0.4 T (inner) and 2.5 T (outer island structure).

Finite element modelling predicted elastic deformations of the coils of ~10 mm at a magnetic field of 2.5 T. The deformation leads to a planarization of the modular field coils which is associated in a small decrease in the rotational transform. As a result the position of the resonances is outward shifted as well. Experiments confirmed the shift of the islands: the 5/6 island was about 10 cm outward shifted for 2.5 T in comparison to 0.4 T corresponding to a decrease of the rotational transform of ꞓ~1%; a value that is slightly smaller than expected from the simulated coil models, figure 23 (right). This benchmark allows optimizing the mechanical coil model which is necessary for the prediction of other magnetic configurations.

Utilizing the trim coils first error field studies were performed as well. Magnetic configurations including a resonance for the most severe B_{11} Fourier harmonic error field component at (ꞓ=5/5=1/1) were not accessible due to technical restrictions in OP 1.1, so a different magnetic configuration with ꞓ=n/m=1/2 at a low magnetic field of 0.3 T was used, since it also amplifies the n=1 intrinsic error field.

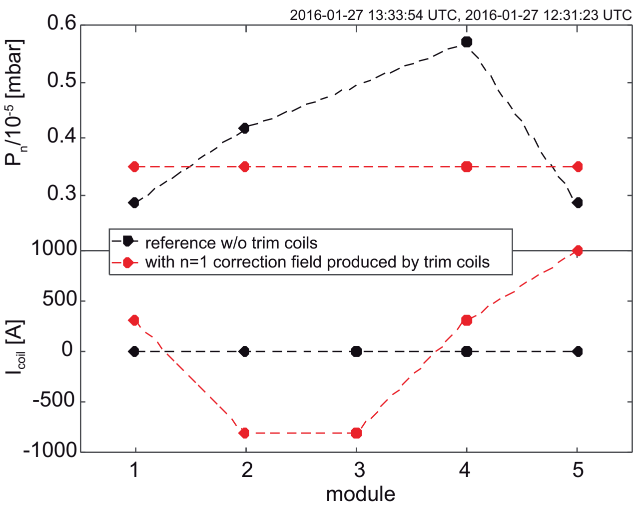

Figure 24: (top) Distribution of neutral pressure measured by neutral gas manometers in 4 of 5 modules. (bottom) Distribution of trim coil currents.

Using an indirect detection technique that applied externally imposed n=1 error fields in addition, it was possible to determine that a ~4 cm wide island (not directly observable) is present in the intrinsic field, and through that, a residual b_{21} component was estimated to ~$5 \cdot 10^{-6}$ which is significantly below the maximum tolerable error. The compensation of that error field is possible by <10% of the nominal current in the trim coils. The suspected main source of the error field, slight misalignment of the superconducting coils, was then confirmed through experimental modelling using the detailed measurements of the coil positions. During plasma operation asymmetries among Wendelstein 7-X modules have been detected by several diagnostics, e.g. neutral gas manometers. The typical distribution during the discharge is shown in figure 24 (top graph, black curve). It was expected that such asymmetries would arise either from field errors or deviations in limiter positions. An n=1 field was added during several experiments utilizing the trim coils. By changing the phase and amplitude of the n=1 field it was possible to find a configuration that reduced the asymmetries in the pressure distribution among the modules. This led however, to stronger asymmetries in the limiter power loads. These observations are still under investigation.

6 Stellarator Dynamics and Transport (E5)

Several diagnostic projects from the former diagnostics subdivision have been regrouped into the experimental division 5 since June 2015. The work in the division was focused on the integration and commissioning of the diagnostic systems to be ready for the first operational phase OP1.1 and to analyze the data recorded during that phase. Furthermore, the development on a set of previously deferred as well as new diagnostic projects have been (re-) started to be ready for the next experimental phase OP 1.2.

6.1 Transport Group

6.1.1 Laser Blow-off

A Laser Blow-Off (LBO) system is presently being designed in cooperation with Consorzio RFX (Padua) for injection of impurity atoms into Wendelstein 7-X plasmas in order to study their transport. It utilizes a pulsed laser beam (20 Hz, 0.9 J) to blow off metal impurity coatings from a glass target.

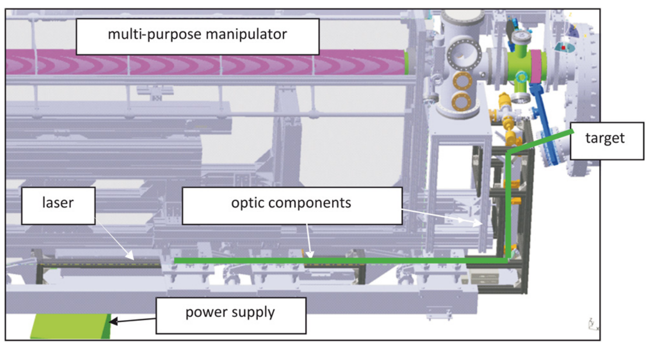

Figure 25: The multi-purpose manipulator is used to insert the glass target into the Wendelstein 7-X vessel to the right position. The laser, the power supply as well as the optic components are installed below the multi-purpose manipulator. A laser beam is guided through two mirrors and two lenses through the entrance port to the glass target.

The target will be positioned in the equatorial plane inside the Wendelstein 7-X port AEK40 by means of the Multi-Purpose Manipulator (collaboration with FZ Jülich). For this purpose the head of the manipulator can temporally be changed in order to carry the targets. Currently, the control system for adjusting the spot size and position of the focused laser beam at the target guided by motorized alignment of lenses and mirrors is under preparation. First glass targets with specific coatings (Cu, Ti, Al) have been prepared by the Leibniz Institute for Plasma Science and Technology (INP, Greifswald). The analysis of the ablation process and the LBO beam quality using these targets will be investigated by the Institute of Physics in Greifswald and the University of Wisconsin in Madison. The system passed the concept design and the purchase of components has started.

6.1.2 HEXOS

The High Efficiency XUV Overview Spectrometer (HEXOS) has successfully been put into operation and was commissioned during OP 1.1. Although two of the spectrometer's four spectral channels were not available due to defective detectors, the spectra of the remaining two channels could be used to identify relevant impurities and monitor the overall plasma impurity content within the accessible wavelength range (20…160 nm). From the time evolution of injected impurity line intensities, a first estimate of the impurity confinement time could be obtained. All defective detectors have been repaired after OP 1.1, and will be mounted to HEXOS for the absolute intensity calibration. The calibration will be conducted after the repair of the motor drive of one of the spectrometer entrance slits, and will allow for absolute impurity content measurements in OP 1.2.

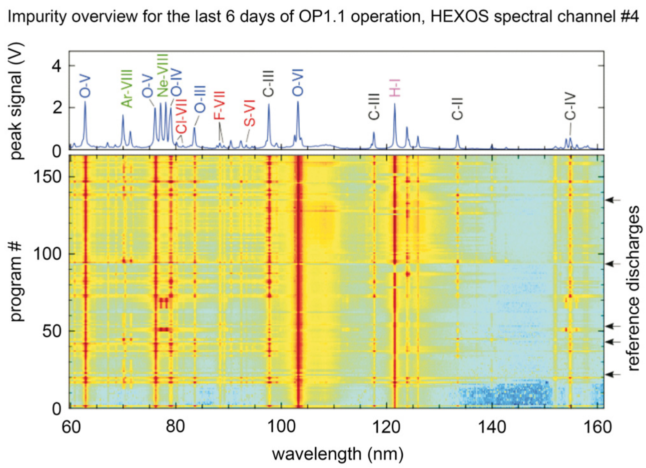

Figure 26: Evolution of the impurity radiation (with respect to the number of experimental program during the last 6 days).

6.1.3 Bolometer

For OP 1.1 a bolometer system consisting of a horizontally and a vertically viewing bolometer camera was installed and commissioned. The horizontal bolometer was used to estimate the total radiated power loss of the whole plasma. The reliability of the results could be proven by global power balance analysis. In combination with the vertical bolometer reconstructions of the 2D emissivity distribution based on gaussian process tomography could be obtained. An increased radiation zone which was typically located in the outer radial region of the confined plasma could be identified. Moreover, a thermal instability, accompanied by poloidally asymmetric emissivity patterns, has been observed during a H-discharge. After the end of OP 1.1 the two bolometer cameras have been removed from Wendelstein 7-X for a hardware upgrade. New detectors will be installed in order to enhance the quality of the measurements in the next experimental campaign.

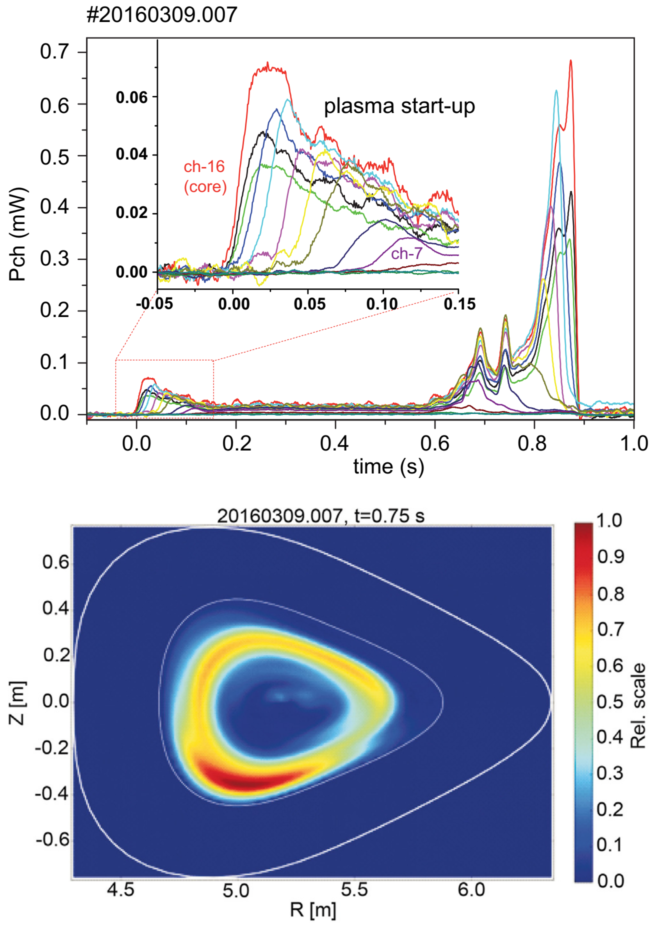

Figure 27: Top: Time traces of the bolometer signals showing the phases of the plasma (20160309.007) going from start-up (t<0.15 s), flat-top (0.15 s < t < 0.57 s), thermal instability (0.57 s < t < 0.8 s) to a radiative collapse (t> 0.8 s). Bottom: Tomographic reconstruction of the 2D-emissivity distribution (#20160309.007 at t= 0.75 s) based on Gaussian Process Tomography showing the poloidally asymmetric emission. The last closed flux surface (LCFS) is marked by the inner white line.

6.1.4 C/O Monitor

The C/O monitor diagnostic, consisting of two double-spectrometers for the detection of a set of light impurity emission lines in the UV range, is dedicated to monitor the content of

Boron, Carbon, Oxygen and Nitrogen in the plasma. The development in collaboration with University Opole and IPPLM (Warsaw) is reaching the detailed design phase. The Wendelstein 7-X flange with the two daughter flanges for both sub-spectrometers has been defined. The diagnostic support structure was divided into two sections for carrying the spectrometers and the pumping system separately. Thereby, only the front part needs to be removed in case of maintenance. The extension and reinforcement of the ground platform has been designed and can be realized in time before OP 1.2. The layout of the output arm permits the mounting of different detector types. In the beginning a charged coupled device (CCD) detector will be used.

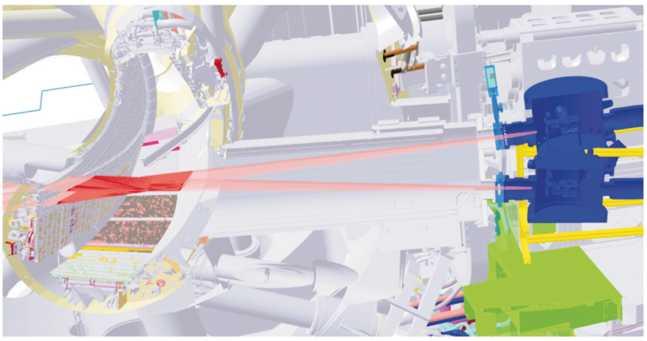

Figure 28: The sight lines of the two sub-spectrometers (each with two spectral channels) intersect the magnetic axis of the plasma. The support structure is not shown.

6.2 Magnetohydrodynamics Group

6.2.1 Magnetic Diagnostics

The complete set of magnetic sensors, namely diamagnetic loops, continuous and segmented Rogowski coils, saddle loops and Mirnov coils have been assembled inside the vacuum vessel and the cryostat. About 50% of the sensors have been equipped with the full set of data acquisition hardware and have been commissioned during OP 1.1. The compensated diamagnetic energy, the plasma current as well as current distributions and selected Mirnov coil fluctuation spectra were automatically recorded. The measurements agree very well with theoretical estimates and could be validated with data from other diagnostics. The diamagnetic loops, saddle loops and Rogowski coils were roughly calibrated by measurements during magnetic field ramps of the external trim coils and comparing to mutual inductances as calculated by the DIAGNO code. Equilibrium reconstructions based on magnetic flux, electron temperature and density measurements have been successfully performed. Towards the next operation phase OP 1.2 the completion of data acquisition hard- and software as well as improvements of data analysis code and equilibrium reconstruction techniques are prepared. An interlock signal for the plasma heating systems based on the plasma energy measurement is being finalized. In addition, a set of 75 temperature sensors inside the vacuum vessel will be taken online. A precise calibration of the continuous Rogowski coils using a current conductor temporally installed in the vacuum vessel is being prepared.

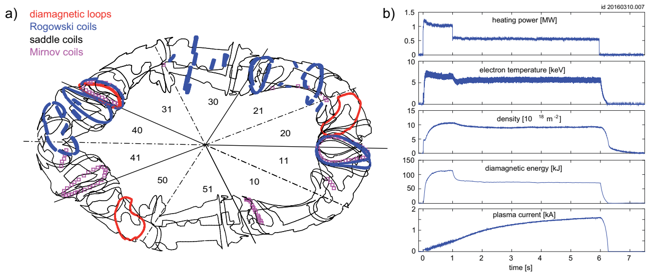

Figure 29: a) Overview of installed magnetic sensors. b) Measurements of the diamagnetic energy and plasma current in addition to the heating power, electron temperature and density during a 6 s hydrogen plasma.

6.2.2 X-ray Pulse Height Analysis System

The pulse height analysis system (PHA) has been assembled and integrated in close collaboration with the IPPLM (Warsaw). After the commissioning of the PHA and the optimization of the diagnostic parameters for the OP 1.1 plasmas, the recorded X-ray spectra could be used to identify intrinsic plasma impurities. In conjunction with the X-ray imaging spectrometers and the HEXOS overview spectrometer, this data helps to understand the impurity composition and transport in the plasma. Since the PHA diagnostic has been installed at a man-access port, it had to be removed from the torus hall after the end of OP 1.1 and is now being upgraded for the next experimental campaign.

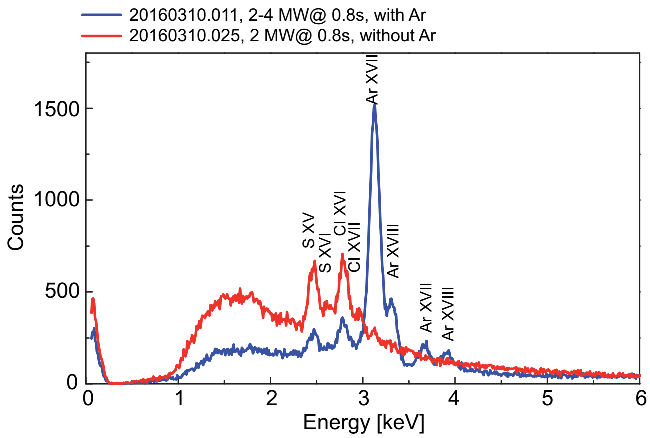

Figure 30: X-ray energy spectra for two experiment programs with (blue) and without Argon seeding (time averaged over the discharge length for improved statistics) showing peaks from impurity line radiation.

6.2.3 Soft-X ray Multi Camera Tomography System (XMCTS)

Restarting the activities on the preparation of the X-ray tomography system in January 2016, the pre-assembly of all four

XMCTS segments, each equipped with five x-ray pinhole cameras, has been completed in June. During this time approximately 350 welding seams have been carried out to build the pipe work systems for the water cooling, the compressed air and the signal cables for the in-vessel electronics.

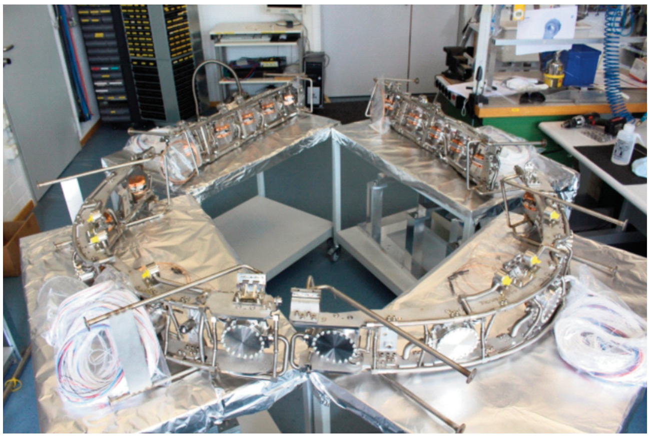

Figure 31: Overview of the four XMCTS segments after pre-assembling in the laboratory.

Special precaution was taken in order to align the camera sight cones with the openings in the attached plasma facing components. After finalization of all welding seams, the preamplifier electronics and the AXUV photo diode arrays have been integrated into the 20 cameras. The electronics of all cameras has been individually tested by illumination of the photodiode array using a modulated light signal fed to the AXUV array by a fibre. The amplitude and phase response is smooth up to 100 kHz. The cut-off frequency is approximately between 200 and 300 kHz. All 400 photodiodes successfully passed the test. Finally, the integral vacuum tests of the pipe systems of all segments have been conducted successfully and the four XMCTS segments were mounted in the plasma vessel in the required positions within the achievable tolerances by the assembly team. All water and compressed air pipe works and three of four signal cable tubes have been confirmed to be UHV vacuum tight. For one signal pipe connection between XMCTS segment and port plug, a small leak (leak rate $1.1 \cdot 10^{-6}$ mbarL/s) has been accepted. The next foreseen steps are the integration of the XMCTS in the Wendelstein 7-X periphery and the diagnostic commissioning before OP 1.2.

6.2.4 Reference Equilibrium Data

Reference equilibrium data for Wendelstein 7-X (calculated with the VMEC code) have been made available by means of web services. A team consisting of members of the CoDaC group, the theory department and E5 is extending the data base as well as the functionality and provides user support. As-built geometries of Wendelstein 7-X coils were analyzed and documented. A collection of coil filament positions for realistic Wendelstein 7-X load cases, including deformations caused by bolts preload, coil self-weight, cool-down to 4 K and electromagnetic load conditions, is being prepared and will be included in the coil data base. This data is used in VMEC equilibrium calculations as well as in vacuum field line calculations for the accurate diagnostic mapping. First equilibrium calculations with real coil geometries showed a 2%-reduction of iota profile. Equilibrium calculations with experimental Thomson scattering profiles were performed for several OP 1.1 discharges. The obtained results are in good agreement with diamagnetic energies measured by the magnetic sensors.

6.3 Turbulence Group

6.3.1 Reflectometry

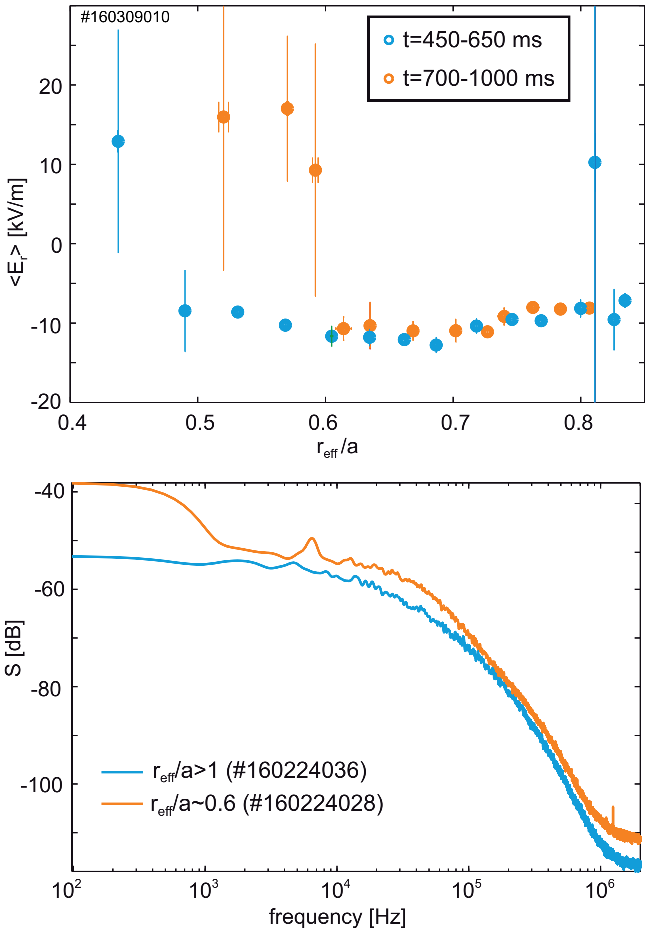

Figure 32. Top: Radial electric field measurements from the poloidal correlation reflectometry (two heating phases, higher heating phase in red colour). Bottom: Density fluctuation spectra inside and outside the last closed magnetic surface.

In the first campaign of Wendelstein 7-X two reflectometry systems have been operated, a Doppler system (cooperation

CIEMAT, Madrid) and a poloidal correlation reflectometry system (cooperation FZ Jülich). Both systems are dedicated to measure the characteristics of density fluctuations and their propagation. From the latter the radial electric E_r field can be deduced. Due to the low plasma densities in OP 1.1 almost entire E_r-profiles could be measured and the transition from the ion to electron root regime could be observed (cf. figure 32). In both heating phases during the same discharge, a transition from negative electric field in the edge to positive electric fields in the core is observed in the radial profile. In the high heating phase the electron root regime (E_r>0) extends to larger radii. Typical measured frequency spectra of density fluctuations in the SOL and core plasma show a broad spectrum up to 100 kHz. For a range of experiments, a peak in the density fluctuations around 7 kHz was observed at a radial position of r_{eff}/a~0.6 (a: minor radius). Possibly, it is related to a global MHD mode forming at the (n,m)=4/5 resonance of the rotational transform ι.

6.3.2 Probe Manipulator

Before the beginning of OP 1.1 the multi-purpose probe manipulator has been installed and successfully operated in close collaboration with FZ Jülich during the first experimental campaign. It allows one to move a probe head over a total range of 350 mm with a maximum acceleration of 3 g. A novel aspect of the probe position system is the use of servo motor drives, which allow the movement and dwell of the probe head at variable positions and velocities. All mechanical, electric and control components of the system worked without problems. A single probe head, provided by FZJ, was used to characterize the magnetic field, plasma profiles and their fluctuations in dedicated experimental campaigns during OP 1.1. The head consists of a set of Langmuir probe pins, which could be used in different configurations with respect to the probe bias, a mach probe arrangement to measure plasma flows and a set of three magnetic pickup coils inside the probe body to measure the magnetic field and its fluctuations.

6.3.3 Phase Contrast Imaging (PCI)

The manufacturing of a phase contrast imaging system in collaboration with MIT (Cambridge/Boston) for the next operation phase has started. It uses a CO_2 laser beam, which is radiated through the plasma to measure the dispersion of plasma density fluctuations along the line of sight. In the advanced scheme used for the current setup, optical filtering allows to localize the fluctuations with regard to the magnetic pitch angle and thereby to assign the measurement volume along the beam path. It will be possible to simultaneously compare the fluctuation characteristics in the core plasma on the high and low field side, where different fluctuation levels are expected. It is planned to compare the data to predictions of trapped electron modes and ITG turbulence.

6.4 Vineta Group

Research on driven magnetic reconnection was further pursued within the framework of the Max-Planck-Princeton Center for Plasma Physics (details are reported in the respective chapter). The second major research objective within the Vineta group is the research into a plasma discharge suitable for the advanced proton-driven particle accelerator project AWAKE, currently under construction at CERN. This activity is embedded into the AWAKE collaboration, lead by CERN and the Max-Planck Institute for Physics. The basic principle is to use a helicon discharge to reach the required high plasma densities. The goal is to develop and install a 10 m long helicon plasma discharge in the AWAKE experiment by 2020 and demonstrate the accelerator potential of the concept by reaching particle energies up to several GeV. It has been demonstrated that the nominal AWAKE plasma density of $n=7\cdot10^{20}$ m^{-3} can be reached in the model helicon discharge with multiple, distributed antenna operation. The necessary RF input power is typically 30kW, which results in an unparalleled power density in the helicon discharge. The temporal evolution of the plasma density is, however, of strongly transient nature, which is attributed to a decrease of the neutral gas inventory (so-called neutral pumping) during a high power discharge pulse. First measurements using a diode-laser based induced fluorescence system indicate that the neutral gas density indeed decreases in the central region of the discharge and that advanced fueling schemes are needed for the stabilization. One requirement for the accelerator scheme is to achieve axial plasma homogeneity better than 10%. This aspect will be the focus of the forthcoming experimental studies.

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ITER

ITER Cooperation Project

Head: Dr. Hans Meister

The IPP contributions to the ITER Project range from R&D for heating systems and diagnostics to the development of integrated control scenarios and theoretical modelling. In addition, IPP is playing a leading role in contributing to the ITER physics through the International Tokamak Physics Activity (ITPA) and by participating in the EUROfusion Workprogramme. Furthermore, IPP participates in European training programmes for young scientists and engineers.

Introduction

In 2015 and 2016 the ITER cooperation project at IPP continued its efforts along the major contributions for the development of heating systems, diagnostics and plasma control as well as theoretical investigations. The ELISE test facility successfully demonstrated the first 1 h pulse in D at low pressures and with an acceptable electron-to-ion ratio. Also, the caesium management in the source could be improved. The physics investigations for the ECRH Upper Launcher investigated various scenarii for the deposition profile width leading to new insights for beam broadening effects. Within the Framework Partnership Agreement (FPA) for the ITER Diagnostic Pressure Gauges the system level design has been completed with the definition of the baseline design. The FPA on the development of the ITER bolometer diagnostic completed the detailed project planning and started system level design activities. In particular, a detailed concept and the required space envelope were developed for all VV mounted bolometer cameras. The work on the Plasma Control System Simulation Platform for ITER released its first version for community use and the plasma control system defined its architecture. Furthermore, as part of contracts with ITER, theoretical investigations for runaway electrons and halo currents have been performed.

Heating Systems

Development of RF Driven Negative Hydrogen Ion Sources for ITER

The two test facilities at IPP, ELISE (equipped with a half-size ITER source) and BATMAN (using the IPP prototype source), are part of the European roadmap towards ITER's NBI systems. The gained experience provides valuable input for commissioning and operation of the Neutral Beam Test Facility (NBTF) presently under construction in Padova, Italy. Particularly, ELISE is part of the annual Work Program in 2015 and 2016. Furthermore, substantial support was given for the design of the RF circuit, the layout of the source and the beam diagnostics.
The aim of ELISE is to demonstrate the ITER requirements with respect to extracted negative ion densities (286 A/m^2 D^-, 329 A/m^2 H^-) at an electron-to-ion ratio below one, a source pressure of 0.3 Pa and a beam homogeneity within better than 90% for 1 h in deuterium and 1000 s in hydrogen. Routine operation for short pulses (20 s plasma with 10 s beam) was demonstrated up to current densities of 176 A/m^2 (D) and 256 A/m^2 (H) at moderate RF power. Linear extrapolation of the RF power suggests that the envisaged ion current densities can be achieved with about 80 kW per driver.

The challenging issue for long pulses at high RF power is the increase of the co-extracted electron current, in particular in deuterium. For suppression of the co-extracted electrons the magnetic filter field and the bias potential applied to the first grid are key parameters, together with the caesium distribution in the source. The filter field is created by a current flowing through the plasma grid; typical currents needed are 4 kA in D and 2.5 kA in H operation. In view of the ITER source it can be stated that the 5 kA installed are sufficient. A significant improvement of the source performance was achieved by strengthening the filter field by external magnets, enabling the first stable 1 h pulse at low pressure in D shown in the figure.

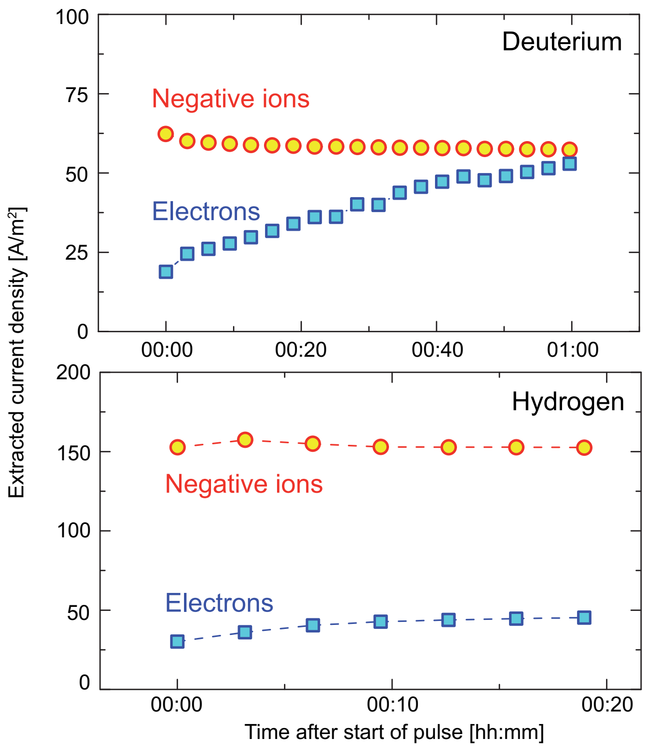

Figure: Extracted negative ion current density and co-extracted electron current density for the best pulses with ITER relevant pressure and pulse length performed up to now at ELISE.

The extracted D^- current was limited to 60 A/m^2 because the extraction voltage and the RF power have been reduced (4 kV and 20 kW per driver) in order to keep the electron-to-ion ratio below one. In hydrogen, the best 1000 s pulse resulted in 150 A/m^2 of extracted ion current being stable within 5%. The low electron-to-ion ratio allowed for 7.5 kV extraction voltage and 40 kW per driver.

Using a dedicated caesium conditioning procedure the long-pulse stability of the co-extracted electrons was significantly improved, particularly in hydrogen. For deuterium, about two times more caesium has to be evaporated. Overall, the caesium consumption has been reduced at least by a factor of four compared to estimations based on the prototype source.
For determination of the beam uniformity data measured at the calorimeter and by beam emission spectroscopy are used. In perveance-matched conditions the beam divergence is about 1°-2° as expected. Deviations of less than 10% in the beam uniformity have been measured in a well-conditioned source.
Further limiting factors for an increase of the RF power were breakdowns at the RF coils as well as overheating of components close to the coils. To mitigate these technical limitations the cooling of the individual components has been optimized and increased gaps between the coils and surrounding components are in preparation.
A thorough inspection of ELISE after two years of operation showed that all components, and in the grids, are in a very good shape. The low caesium consumption resulted in a low caesium contamination of the source. However some unexpected features as copper sputtering between the grids and black rings around the apertures of the second grid have been observed which need further investigations.
The prototype source at BATMAN was re-installed into the standard configuration concerning position and strength of the magnetic filter field. A low amount of co-extracted electrons, as known from previous campaigns, was achieved in this configuration. The position of the Cs oven has no influence on the source performance which confirms that the plasma redistribution of Cs is the dominant process. Dedicated experiments have been carried out at BATMAN to support the DEMO NBI activities (see chapter 5). A new ITER-like grid system for BATMAN has been designed and is under construction to upgrade the test facility. A repeller electrode upstream of the grounded grid can optionally be installed which is positively charged against ground to affect the onset of the space charge compensation and to reduce the backstreaming of positive ions. For magnetic filter field studies a plasma grid current up to 3 kA will be available as well as permanent magnets embedded into a diagnostic flange or in an external magnet frame. BATMAN upgrade is foreseen to go into operation in 2017.

Design of the ICRF Antenna for ITER

Though the framework contract was signed with F4E by CCFE on behalf of the consortium partners, little work has been done, as what needed to be done was being negotiated between F4E and IO.
Since additional neutron shielding is required around the antenna (as it is for all ports), an update of the antenna design would be necessary, but the question is whether, in view of the revised timescale of ITER and the definitions of the interfaces, it may not be better to concentrate on R&D first. Whereas IPP plans to keep involved in this work at the level of 1 ppy, no manpower has been available since the engineer working on this left for industry and was not replaced.

Upper Launcher for Electron Cyclotron Waves

The ITER upper launcher (UL) for electron-cyclotron (EC) waves is being developed by the ECHUL Consortium, which involves KIT, CRPP/SPC, CNR-IFP, DIFFER and IPP in the frame of Grant 615. The main physical goal of the UL is the control of MHD instabilities. The involvement of IPP has the objective of assessing and guiding the design of the launcher through an analysis of its physics performance, which results in a constant iteration between physics and design activities.
The issue of the optimum ECCD-profile width for NTM stabilization has been addressed, in particular to guide the design of the mirrors and clarify the needs in terms of multi-beam overlap. The analysis is based on the criteria for NTM suppression: Efficient NTM control is achieved when the deposition is within the expected marginal island width; once the deposition profile falls below this width, the gain from further focusing is small. The beams originating from the lower steering mirror are in this sense slightly over-focused, so that two optimization paths can be addressed. The design constraints on the mirrors may be relaxed by allowing the width of the beam waist to increase by about 20% when keeping the present toroidal injection angle fixed. On the other hand, a larger launch angle could be exploited to increase the total driven current, while leaving the deposition profile narrow enough. However, because of the present uncertainty about the “effective” profile width, the latter option is not recommended. In particular, the effect of beam broadening due to density fluctuations at the plasma edge could significantly spoil the expected ECCD localization in machines of ITER size.
This last issue has been tackled by means of a comprehensive numerical approach, based on the solution of the wave kinetic equation and using a perturbative treatment of the scattering term (“Born approximation”), implemented in the wave-kinetic-equation solver WKBeam. A significant effort has been devoted to the validation of this model by means of a comparison with full-wave simulations, see section on Wave Physics in chapter 7. The Born approximation appears to be well suited for these investigations. Quantitatively, the expected broadening depends critically on the fluctuation profile, for which a standard scenario has been established on the basis of results from present tokamak experiments. For H-mode plasmas, the region between the minimum in the E_r-well and the near SOL is predicted to dominate the scattering of EC waves. This is the starting point for extensive parameter scans which have now been started. First results concerning the scattering from O- to X-mode have been obtained in the frame of a theoretical development of the WKBeam code coordinated by the department for Numerical Methods for Plasma Physics.

Diagnostics

ITER Bolometer Diagnostic

The activities within the ITERBolo project continued in 2015 with the further planning of the FPA. Based on the analysed documentation a full work plan was set up covering all activities to be performed within the foreseen duration of the FPA. Work packages were defined as well as the corresponding resource loaded schedule for sensor development, prototype testing and design integration in ITER. However, the proposed activities started slowly and delayed because of a thorough re-assessment of the budget and overall schedule for ITER being performed during 2015 and until mid-2016 at the ITER Organisation and within the Domestic Agencies, which also led to a delayed start of the subsequent Grant on system level design activities and the necessity to re-evaluate the planning. This will now be done with the new milestone for the ITER first plasma in mind, which is set for 2025.

The design integration activities made substantial use of previously gained results and focused on the development of a detailed conceptual design for the bolometer cameras to be placed on the ITER vacuum vessel (VV) wall. As blanket modules (BM) will be installed only after first plasma and may be aligned in their position by ±15 mm in both, toroidal and poloidal direction, the camera is separated in two parts: The mounting platform fixes the signal cables and is installed before first plasma. The main camera will be installed after first plasma and after customisation of the collimator position and the 3D-PCB supporting and electrically connecting the sensor to match the finally defined positions of the BMs. The material choice of the platform and housing is currently being investigated. As no active cooling will be available for the VV cameras, a good thermal connection to the VV wall acting as heat sink is essential and realised by a copper mesh beneath the main camera body. Furthermore, the design aims at providing a direct heat flow path from sensor to VV wall and from collimator to VV wall, but separating those two as much as possible. Current analyses estimate the temperature at the sensor to be in the order of 220 °C. These design activities led to the definition of the required space envelopes which are essential for finalising the interface definition with the BMs as those are about to go into manufacturing.

Bolometer cameras in the divertor cassettes have to be placed so that they can observe the plasma through the gaps between adjacent cassettes. This subjects them to requirements imposed by the remote-handling installation process of the divertor cassettes resulting in strongly reduced space envelopes available for the cameras. Furthermore, it implies an asymmetric heat flow path. To avoid strong asymmetries in the resulting temperature pattern for the bolometer sensor, a dedicated parametric model of the camera design has been implemented and analysed. For various locations it could be demonstrated that design parameters can be optimised so that the temperature gradient between measurement and reference absorber is reduced to below 1 °C and thus minimising the impact of temperature gradients on the measurement error.

For port plug cameras scoping studies were made to determine the type and the main design parameters of the cameras. Using the LOS distribution as proposed during the conceptual design review, the required space envelopes have been defined in cooperation with the port integrator. In the equatorial port plug EPP01 there are three cameras with 90 LOS. Both upper port plugs equipped with bolometer cameras (UPP01 and UPP17) have identical space envelopes and camera types, but differ slightly in the orientation of the LOS. In each of them are located 70 LOS.

To support the reliable definition of bolometer performance the novel library package ToFu dedicated to tomographic reconstruction methods in 3D geometry has been implemented for the use of the ITERBolo project. Using the geometries of sensor and aperture coordinates as foreseen for ITER together with plasma radiation profiles simulated for the ITER standard scenario led to the estimation of power levels expected for the individual LOS. Currently, these routines are being used to estimate the loads from plasma radiation onto the diagnostic components.

Most prototype tests proposed will be carried out during later Grants, once reliable designs for components are available. So far, input was provided to specify irradiation tests and to support the manufacturing of simplified 3D-PCBs. The experience gained for the latter is essential for the design development. Furthermore, tests on prototypes of in-vessel cables for ITER started.

The sensor development together with F4E is significantly delayed due to the delays in placing the respective contracts after the re-scheduling at F4E. Meanwhile, IPP cooperated with Fraunhofer-IMM to produce the first prototypes of bolometer sensors based on flexure hinges, an option not foreseen to be followed by F4E. They showed very good mechanical stability during repeated temperature cycles up to 450 °C. First measurements proved that they can be used as bolometer sensors, albeit with increased cooling time constants and heat capacity. First samples will be installed in ASDEX Upgrade and demonstrate their applicability in a fusion experiment. Additionally, bolometer measurements which detect the voltage and current on the measurement circuit simultaneously could be used to define a procedure for compensating the temperature drift of calibration parameters and thus reducing measurement errors at varying operating temperatures.

ITER Diagnostic Pressure Gauges

During the present reporting period two Grants with F4E (SG03: System level design & further design of sub-systems and SG04: Engineering analyses & performance modelling) have been completed successfully. SG04 was executed in collaboration with the engineering company Sgenia that has, at the end of it, left the DPG Consortium. All objectives have been achieved and all deliverables have been submitted to F4E.

The first part of SG03 was dedicated to System Level Design. At the beginning of 2015 three architecture options for the DPG head and five for DPG electronics have been proposed. A set of criteria apt to evaluate, rank and select the different options were defined. The most promising option found for the gauge head is a hot-cathode ionization gauge of ASDEX type with a thermocouple to monitor the temperature in the gauge head for accurate calibrations. For the electronics the option with highest ranking is the one which accommodates all components in the diagnostic building for easy access. Functional and risk analyses were carried out leading to the identification of 130 risks and a mitigation plan for the critical ones. According to the results of the RAMI analysis, the configuration of the DPG system, when any 8 of 52 gauges are in operation, has a mean availability of 99.9%, which is higher than the 98% requirement for the pressure measurement availability. The system requirements were extracted, in some cases reformulated, and implemented in a specific module within DOORS software. Irradiation effects and resulting structural and operational degradation of the DPG head have been investigated component-by-component. Analysis of IPP and simulations performed by F4E showed that heating rates, DPAs, gas production and activity in different materials of the sensor head expected for the ITER environment will not affect operation of the DPG sensor. Accordingly, the benefit/cost ratio of irradiation tests of DPG components is low and therefore they are not recommended.

Physical processes that lead to slow deterioration (long term effects) of the filament have been identified. The following long-term wear effects with a potential negative impact on the filament have been analysed and assessed: Fatigue, creep, material evaporation and exhausting of dopant. Results of the assessments demonstrate that these effects will not appear or not affect operation considering the foreseen 1500 h averaged operation time per gauge head.

Sensor head preparatory design activities resulted in the design optimization of ceramic insulators and housing, including the baffle and the support structure. The thermocouple type K has been recommended as suitable for ITER. Several proposals for the filament fixation system, the connection of cable terminations with grid pins as well as for the connection of cable connectors to the support structure have been given.

Furthermore, the following cable tests have been executed: i) Long cable tests, ii) EMC tests of an ITER ex-vessel cable loom and iii) ITER ex-vessel cable loom tests at ASDEX Upgrade. The long cable tests demonstrated the possibility of the DPG head to operate with transmission lines up to 200 m. In- and ex-vessel cables have been recommended and first versions of cable diagrams and cabling list have been produced. For the electronics components-of-the-shelf solutions for filament heating and electrodes potential power supplies as well as for thermocouple electronics have been recommended. Individual designs were proposed for electron and ion current measurements and control electrode switch. The principal scheme of the filament heating control loop has been developed. The preparatory design of signals routing between frontend electronics and the plant system controller has been done. A proposal for how to integrate DPG electronic components into a cubicle has been given.

The testing strategy has been defined and divided into three tasks: i) Measurement performance, ii) Reliability performance and iii) Temperature measurements, all leading to the creation of the technical specifications for the next Grant (SG05).

SG04 performed the FEM analysis of the whole DPG. Among the main findings are: i) A limitation in the heating current/ambient B-field operation due to high stresses on the filament with present design and materials (short term effects). ii) High stresses in the filament fixation region which could lead to failure of the system. These two subcomponents must be carefully considered before final design.

Another important task dealt with the DPG measurement performance simulation and improvement. The IPP DPG simulation code was parallelised decreasing run-time by typically an order of magnitude. Using this new parallel implementation, a wide parametric study has been achieved to optimise performances in terms of pressure range and accuracy. The simulated saturation region was increased from 2-3 Pa up to 15-20 Pa thus close to the ITER requirement. Identified key parameters are the grid transparency and the potentials. The grid transparency has been set to 40%, compromising between saturation limit, signal strength and manufacturing feasibility. Effects of E×B drifts have been included in a phenomenological approach. It has been demonstrated that the electronic scattering is a dominant phenomenon and with the present implementation cannot be correctly addressed and thus an experimental validation of the results is critical.

The next project phase (SG05) deals with the testing and validation of engineering, life-time and measurement performances of the current design and is foreseen to be started in November 2016.

Control and Data Acquisition (CODAC)

Control of the ITER plasma will be a much more demanding task than that of today's Tokamaks as it requires developing complex schemes which can control competing objectives with limited actuator authority in the vicinity of stability limits. As a consequence, much more integrated, multivariable and nonlinear control and actuator management schemes must be applied, and more comprehensive exception handling must be devised to reliably manage deviations and failures in real-time. IPP supports two major activities for the development of the ITER plasma control: The Plasma Control System (PCS) and the Plasma Control System Simulation Platform (PCSSP).

PCSSP was jointly developed for ITER since 2010 by a consortium of IPP with GA (US, lead) and CREATE (I). It will

specifically assist in four key tasks: development and test of the ITER PCS architecture, development and test of ITER control algorithms, implementation and test of real-time code, and deployment in ITER operation.
To address these issues PCSSP connects a PCS model with a Tokamak simulator in open or closed loop topology, to observe how these interact. The PCS model can connect directly or through simulated networks to the Tokamak simulator. The Tokamak simulator may include models of plant systems, such as actuator and diagnostic modules, and of plasma modules for computation of physics aspects. The choice of modules depends on the required purpose and level of detail, and might include fast-running control-oriented models, but also connect to external detailed physics-oriented models. Simulations are driven by input data from local test data for specific modules, or from a central Pulse Schedule file, which can describe the planned evolution of a complete ITER pulse. An event generator produces asynchronous test information to simulate the occurrence of technical or physics situations, which require control response, to study handling options and intervention methods, and how these can best be implemented in the PCS architecture.
PCSSP is based on industrial technology (Matlab/Simulink) so that developers can build on rich evaluation and control related function libraries and state-machines, and use graphical interfaces or scripting and animation tools for fast progress. A first PCSSP version was released for free use in the ITER community (https://git.iter.org/projects/PCS/repos/pcssp). It comes with a set of PCSSP service functions (configuration, I/O), control functions (FB controller for shape, density, ECCD; Supervision Controller, quality tagged data exchange), diagnostics and actuators (PF, NBI, EC, gas puff), and various modules for dedicated physics characteristics (magnetics, kinetics, NTM, EC beam propagation), plus a reference generator module to provide Pulse Schedule data. More modules are expected to be developed by users of PCSSP in a joint effort, in particular during the PCS control analysis and architecture development. PCSSP is also the candidate platform for implementing an ASDEX Upgrade flight simulator tool. For this purpose, dedicated PCSSP plugins, modelling AUG physics and components will be developed.
PCS is developed by the ITER team in cooperation with a consortium of IPP (lead), GA (US), CREATE (I), CEA (F) and CCFE (GB). The focus of the actual Preliminary Design phase which ends with the design review in November 2016 is the 1st plasma and the early plasma operation in H/He up to 15 MA. As work method, a physics team analyses how the plasma can be controlled and describes the control scenarios, while an architecture team derives a system engineering model, puts it to a control database and examines dependencies among control and Exception Handling functions to refine the PCS design. Critical issues for control of the initial ITER plasma are the breakdown in the large vessel in presence of eddy currents, the risk to produce runaway electrons with ohmic plasmas at low neutral pressure, the control of the prefill neutral pressure during plasma initiation, the feedback control during current rise at I_p<1 MA with challenging signal-to-noise ratio of magnetic data, the avoidance of unabsorbed EC power in the vessel, density control delays caused by long gas puff tubes, and the detection and handling of disruptions and runaways. In addition to dedicated control analysis, scenario descriptions for the nominal pulse phase chain from ignition to flat-top, investigated controlled and hard termination schemes and the role of PCS assisting the Central Interlock System in investment protection have been assessed. Those descriptions of how control functions must be combined and sequenced in different phases of a pulse are essential for an improved design of control schemes and exception handling mechanisms in PCS.
The resulting preliminary architecture of ITER PCS will consist of two layers: A Pulse Continuous Control (PCC) layer operates control loops with distributed processes for data measuring and evaluation, feedback and feed-forward control, and the management and control of actuators. Logically on top of PCC is the Pulse Supervision Control (PSC) layer, which orchestrates PCC activities and executes the Pulse Schedule. Between these two layers complementary Exception Handling functionalities can be deployed: PCC implements local Exception Handling to modify control behaviour (substitute input data, adapt control parameters, change algorithm) or to modify control schemes (change references or interaction among controllers) in case of controller degradation. PSC implements Exception Handling to modify the control goal (change pulse segment or investigation) in case the pulse must be terminated or the nominal pulse path be replaced by an alternate.

Plasma Theory for ITER

Theory for Runaway Electrons in ITER

A consistent description of runaway electron (RE) generation in plasma disruptions remains an open physics issue. The possibility of the formation of high post-disruption RE currents raises safety-related concerns for large tokamaks, such as ITER. Although the avalanche mechanism of RE production is anticipated to be the dominant mechanism in ITER [M. N. Rosenbluth, S. V. Putvinski, Nucl. Fusion 37, 1355 (1997)], the avalanche multiplication of the runaways after the thermal quench still requires a seed RE current. The need for reliable prediction of the RE generation in ITER calls for additional attention to the primary (seed) population of the RE.
A systematic description of electron kinetics during impurity-dominated thermal quenches in tokamaks was developed. A 2D Fokker-Planck equation for the hot electrons and a power balance equation for the bulk plasma are solved self-consistently, with impurity radiation as the dominant energy loss mechanism.

The post-quench RE density, energy and current are found for a broad range of initial plasma parameters, including those of interest for ITER. A prompt conversion of the total pre-quench current into the sub-MeV RE current is feasible in the case of abundant impurity injection. The subsequent decay of such RE current should be governed by the near-threshold regime [P. Aleynikov, B. N. Breizman, PRL 114, 155001 (2015)] and prescribed by the impurity amount.
It is found that the runaway seed density is a non-monotonic function of the pre-quench plasma temperature and that the seed current tends to be restrictively small in plasmas with high pre-quench temperatures, which is likely to cause non-monotonic seed RE profiles in ITER and future high-temperature tokamaks. The non-uniformity of the plasma creates a possibility for the post-quench current to be carried by two distinct runaway populations (a sub-MeV and an ultra-relativistic).

Modelling of Halo Currents

Within an ITER-Contract and complementing EUROfusion activities, methods have been developed for extending the JOREK-STARWALL non-linear MHD models to Halo currents in collaboration with the Romanian association and ITER Organization. Previously the non-linear MHD code JOREK and resistive wall code STARWALL were coupled in a fully implicit way in order to describe the interaction of the MHD activity in the plasma with eddy currents induced in 3D conducting structures surrounding it. This has been applied, e.g., to resistive wall modes, vertical displacement events and QH-Mode. The extension of the JOREK-STARWALL models to Halo currents, which is of particular interest to ITER because of the associated rotating non-axisymmetric forces occurring during disruptions, requires an extension of STARWALL and of the coupling between the codes. Three approaches for extending STARWALL, in which conducting structures are represented by thin triangles, to Halo currents were considered and the most efficient approach was implemented in a small production-like code and validated against the other approaches as well as analytical theory for 3D wall geometries with holes. The application to a simplified ITER wall geometry was demonstrated. The Halo current code allows calculating the current distribution in the conducting structures for a given "source/sink" distribution of the wall currents, i.e., distribution of currents flowing across the plasma-wall interface. For this purpose, a "response matrix" is calculated that allows determining the current distribution in the wall by matrix vector multiplication with the current source/sink distribution. The coupling of this code to JOREK-STARWALL is foreseen within the framework of a follow-up project. For this coupling, the response matrix will be used inside JOREK to express the Halo current evolution in terms of JOREK variables at the boundary of the JOREK computational domain in a way that conserves the full-implicitness of the extended JOREK-STARWALL.

Axisymmetric walls were implemented directly inside the JOREK computational grid for testing purposes. However, the implementation turned out to be significantly more challenging than expected, for instance due to the incompatibility of the boundary conditions between plasma and wall with the C1 continuous finite element grid structure of JOREK. For this reason, the implementation has not been completed and will not be used as a testbed for the JOREK-STARWALL with Halo currents like it was originally foreseen. Instead, the Halo current extension will be benchmarked with other codes and analytical models when it is finished.

Advancement of Young Scientists

IPP is strongly involved in FUSENET (European Fusion Education Network), among others through membership in the Board of Governors and the Academic Council. In 2015, five fellowships have been granted in the fourth cohort of the Fusion-DC (Erasmus Mundus funded) programme and, in 2016, six fellowships in the fifth cohort.
Five new students (3 in the 4th cohort and 2 in the 5th) are doing most of their PhD research work at IPP. In two cases, IPP is the home institute. One student of the first cohort, who did part of her work at IPP obtained her PhD degree in co-tutelle LMU-UGent, and one student with a point Padua-Lisbon PhD degree started as a Post-Doc at IPP. This international programme has been very highly commented by the commission that evaluated the European doctoral programmes. The Human Resource Report recommended to continue funding the programme beyond the Erasmus Mundus funding.

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DEMO

DEMO Design Activities

Head: Prof. Dr. Hartmut Zohm

The 'DEMO Design Activities' project focuses on aspects of physics and technology relevant for tokamak and stellarator designs, in line with the unique position of IPP following both lines. Many of the activities are carried out under the EUROfusion PPP&T Programme, where substantial EU collaborations exist. On the national level, the German DEMO Working Group joining scientists from FZJ, IPP and KIT serves to strengthen collaboration and strategic planning.

Design of a Tokamak DEMO and its Operational Scenario

Studies of the design point of a tokamak DEMO were conducted based on a 'stepladder' approach, defining an attractive Fusion Power Plant (FPP) and then scaling down to DEMO, fixing the parameters A, H, β_N, f_{GW} and the absolute value of n_e. This choice should make sure that the operational scenario is similar between DEMO and the FPP, and only R and B_t are changed. Inserting the ITER values of R and B_t, it can be evaluated how this scenario can be prepared in ITER. For A=3.1, H=1.2, β_N=3.5, f_{GW}=1.2 and n_e=8×10^{19} m^{-3}, an attractive steady-state stepladder is found, albeit with the window $P_{LH} < P_{sep} < P_{max}$, exhaust decreasing with machine size. 1.5 D ASTRA simulations largely confirm these findings.

Power exhaust continues to be one of the decisive elements in tokamak DEMO designs, requiring operation with core impurity seeding to reduce P_{sep} to an acceptable level. Previous studies (see Annual Report 2014) have shown that Xe would be an effective radiator with low impact on confinement.

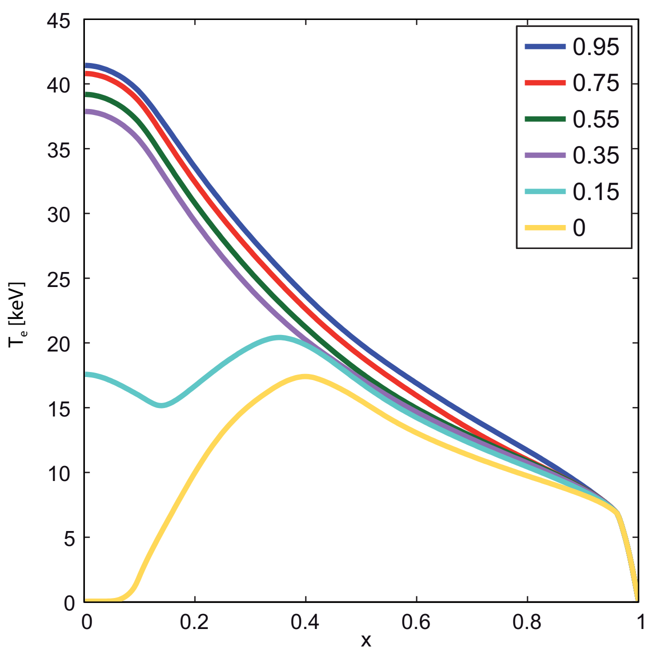

Figure 1: Using TGLF, radiation losses due to impurity seeding (modelled as Gaussian with width x=r/a=0.1) significantly affect the temperature only if located within x=0.5, i.e. when overlapping with the α-heating.

A study using ASTRA and TGLF has now shown that this ability crucially depends on the region affected by the radiation losses, in agreement with experimental observations: if the bulk of the impurity radiation is located outside the region heated by α-particles, temperature profile stiffness ensures little impact on the core temperature while for central radiation, the temperature profile collapses (see figure 1). This is in line with results from impurity seeding experiments on ASDEX Upgrade.

A new 0-D model for SOL and divertor properties in the presence of impurity seeding was developed and benchmarked against ASDEX Upgrade SOLPS runs. It captures the transition from conduction to convection and should give an indication of the threshold for divertor detachment. This model has already been coupled to ASTRA to evaluate two-species impurity seeding for simultaneous adjust of P_{target} and P_{sep}, albeit presently not in a time-dependent manner.

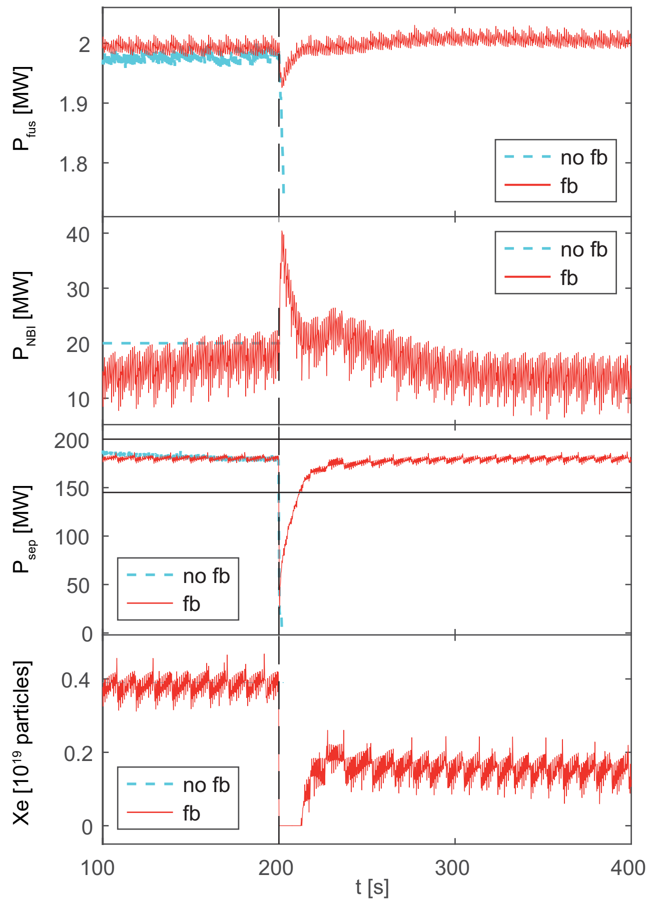

Figure 2: Simulation of control of DEMO operation: a W-flake penetrating into the plasma leads to a collapse if no control action is taken (blue). Using simultaneous control of impurity seeding and NBI heating, the perturbation can be recovered (red).

First simulations of simultaneously controlling P_{fus} and P_{sep} in DEMO were carried out using a new environment where ASTRA has been embedded into the Simulink control suite. Figure 2 shows an example where a W-flake of mass 3 mg is introduced into the plasma as perturbation. Without control, the increase in radiation leads to a loss of the plasma (blue traces). With control (red traces), the P_{sep} controller shuts off the Xe puff and the P_{fus} controller increases P_{NBI} to recover the plasma operational point. In the future, also multi-input multi-output controllers will be simulated and inclusion of noise and latencies will help determining realistic control margins.

Studies on a Next-step Stellarator

In order to bridge the gap in physics and technology from W7-X to a stellarator fusion power plant, an intermediate next-step stellarator is a prudent approach to reduce development risks. Two different variants of next-step stellarators based on the HELIAS concept were studied. A device targeted at burning plasma physics, i.e. ITER-like, could be built using the W7-X magnet technology (NbTi) and would sit at R=14 m, while a T-self sufficient DEMO-like device that would also generate electricity would largely benefit from higher field, i.e. the use of Nb_3Sn, and sit at R=18 m (see figure 3). This comes from the fact that the lines of constant β roughly coincide with Q=const. in figure 3.

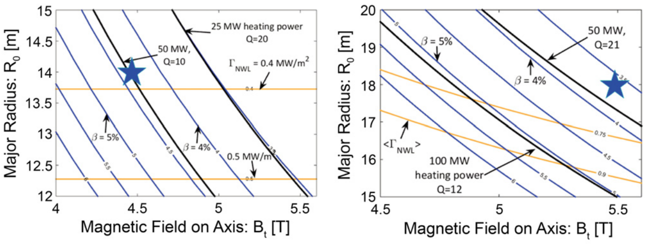

Figure 3: Options for next-step HELIAS stellarators: ITER-like design with Q=10, no T breeding, no electricity generation (left) and DEMO-like design Q=20, P_{el}=200 MW, T-self sufficient (right).

Using the stellarator modules implemented in the PROCESS code (see Annual Report 2014), a costing of a tokmak (R=8.5 m) and a HELIAS (R=22) DEMO was done, showing that the costs will not differ greatly. Quite surprisingly, the cost for the magnet systems is about equal (due to the large cost of the poloidal field coil system of the tokamak), and the higher cost of the tokamak H&CD systems needed for current drive roughly outweighs the higher cost of the HELIAS blanket due to the increased complexity in 3-D.

Technology Studies for DEMO

In the area of H&CD systems, the impact of these systems on the Tritium Breeding Ratio (TBR) was determined in collaboration with KIT. Reasonable specifications of a neutral beam injection (NBI) system were estimated for a pulsed DEMO, based on the assumption of a moderately evolved ITER NBI design delivering 50 MW from two beamlines. Beamline length, beamlet divergence and injection geometry were found to be the major factors determining the required port size, yielding a reduction in TBR in the range 0.4 - 1.1%. A focus of the development is laid on reducing the complexity of the beam source while at the same time improving the system efficiency, e.g. by development of a laser neutralizer (see also the contribution by University of Augsburg in this scientific report). In the field of ICRH for DEMO, work concentrated on the study of the distributed antenna concept, where the impact of such an antenna on the TBR has been evaluated to be in the range 0.35 - 0.6%. Also, first steps in refining the options on the feeding of the antenna were carried out. Finally, a preliminary estimate of the TBR reduction due to an ECCD system in DEMO was of the order 0.13%.

IPP is project leader for the EUROfusion work package "DEMO divertor", which consists of two subproject areas "Cassette Design and Integration" and "Target Technology". Recently, various physical loads have been assessed for structural design study and the first thermo-hydraulic design of cooling schemes was defined for two different coolant media (water and helium). Advanced target design concepts with novel heat sink materials and joining technologies are being developed.

Finally, IPP is strongly involved in the conceptual studies of the gas injection system employed for pre-fill, ramp up and plasma seeding as well as the pellet core fuelling systems for DEMO. The pellet guiding system needed for efficient injection from the torus inboard side turned out to be a key element in the fuel cycle. Optimization its design is expected to result in a significant reduction of the required T inventory. Supportive experiments to engineer a well matched pellet transfer are performed employing the ASDEX Upgrade pellet launching system.

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Plasma-Wall Interactions and Materials

Plasma-Wall Interaction

Head: Prof. Dr. Rudolf Neu

Within the project "Plasma-Wall Interaction" the areas of plasma-surface-interaction studies, material modification under plasma exposure, development of new plasma-facing materials and their characterisation have been merged to form a field of competence at IPP. The work supports exploration and further development of the fusion devices at IPP and also generates basic expertise with regard to PFC-related questions in ITER, DEMO and future fusion reactors.

Surface Processes on Plasma-Exposed Materials

Sputtering of Iron, Chromium and Tungsten by Energetic Deuterium Ion Bombardment

Reduced-activation ferritic-martensitic (RAFM) steels, such as EUROFER, which are being developed as structural materials for fusion applications, are recently also considered as a possible option for certain areas of plasma-facing surfaces in a future power plant. Sputtering of RAFM steel is more complex than for pure elements because steel is a compound material. For example, one can theoretically expect that lighter alloyed elements will be preferentially sputtered, leading to a continuous change of the surface stoichiometry during ion irradiation until a steady state is reached. For a better understanding of the sputtering processes on RAFM steels it is in a first step necessary to know the sputtering of the most important alloying elements as a reference. However, sputtering data for these elements are still quite limited.

Therefore, sputtering of the pure materials iron (Fe), chromium (Cr) and tungsten (W) due to energetic deuterium (D) ion bombardment was investigated. Sputtering yields were measured as a function of the D ion energy from 60 to 2000 eV/D. The obtained data can be well reproduced by a semi-empirical formula suggested by Bohdansky, and the corresponding fitting parameters were determined. It is further confirmed that analytical formulae suggested by Eckstein and Yamamura agree satisfactorily with these experimental data. By comparison with results from the binary-collision-approximation-based calculation codes SDTrimSP and SRIM it is found that SRIM has some limitations in simulating sputter yields close to the threshold whereas SDTrimSP results show good agreement with measured data in the investigated energy range. These data form the basis for future investigations on the sputtering of RAFM steels.

Optimization of the Depth Resolution for Deuterium Depth Profiling up to Large Depths

The nuclear reaction $D(^3He,p)\alpha$ is generally used to measure the deuterium (D) depth profile of D in plasma exposed materials. Based on general kinematic considerations it has been shown that the depth resolution for D depth profiling using this nuclear reaction is optimal at reaction angles of 0° and 180° at all incident energies below 9 MeV and for all depths and materials. In order to confirm this theoretical prediction the depth resolution was determined experimentally with a conventional detector at 135° and an annular detector at 175.9°. D-containing thin films buried under different metal cover layers of aluminium, molybdenum and tungsten with thicknesses in the range of 0.5-11 µm served as samples. For all materials and depths an improvement of the depth resolution with the detector at 175.9° is achieved. For tungsten as cover layer an improvement of the depth resolution by a factor of up to 18 was determined. Good agreement between the experimental results and the simulations for the depth resolution was demonstrated.

Migration of Materials in Fusion Devices

Experiments with the New Divertor Manipulator System DIM-II in ASDEX Upgrade

The outer divertor of ASDEX Upgrade is equipped with a unique manipulator system (DIM-II), which allows exposing two adjacent test target tiles at the central position of outer divertor module S2. Samples can be exchanged in between experiment days by means of an airlock system without breaking the vacuum of the ASDEX Upgrade vessel. The default probe head for exposure of standard geometry target tiles was extended in 2015 to allow installation of instrumented tiles equipped with electrical probes and thermocouples. In addition a new probe head was commissioned, which provides a rotating sample exchange mechanism. This system allows exposure of small samples flush with the tile surface for individual discharges with sample exchange in between. As a further extension of the manipulator systems an active tile cooling system was procured, which for the first time will allow dedicated tests of actively cooled target component concepts under reactor-relevant thermal load conditions and realistic magnetic field geometry.

The probe head with instrumented tiles was used for a study of power handling capabilities of tungsten divertor target plates under repetitive transient ELM power loads. At two poloidal positions thermally and electrically insulated massive tungsten samples with protruding surfaces were installed. One sample provided a leading edge while the other sample provided a sloped face. During strike point excursions to these surfaces both the current flowing to vessel potential and the integral power transfer to the samples were measured, the latter by thermo-couples attached to the bottom side of the samples. The main objective of the experiment was to verify that the power load to a surface structure is indeed given by the geometric projection of the parallel power flux along the magnetic field lines, which usually is derived from infrared thermography measurements. Data from previous experiments at the JET tokamak suggested violation of this model.

In the ASDEX Upgrade experiment no discrepancy of directly measured heat input and the corresponding value derived from geometric projection of incident parallel power flux was observed. Indeed, meanwhile the discrepancies observed at JET could be rectified by a more detailed interpretation model. During the probe exposures analysis of the current flowing to vessel potential allowed for the first time to quantify the thermionic emission current connected to transient tungsten melting during ELMs. Post exposure analysis of the molten samples showed indeed a pattern of re-solidified melt debris as predicted by simulations of melt motion during repeated transient melt events.

Effect of the Temperature on the Surface Morphology, D Retention and Erosion of EUROFER Exposed to Low-Energy High-Flux D Plasma

In recent years the use of a bare RAFM steel wall without a protective amour has been proposed as plasma-facing material (PFM) due to technological and economical reasons (see above). Therefore, the erosion and hydrogen retention of the RAFM steel EUROFER was measured at high particle fluxes. It is expected that the small concentration of W (and Ta) contained in EUROFER enriches at the surface due to preferential sputtering and, therefore, mitigates the unacceptable high erosion of Fe. EUFROFER samples were exposed in the Pilot-PSI device at DIFFER to deuterium (D) plasma with incident ion energy of ~40 eV and incident D flux of 2-6×10^{23} D/m^{2}s up to a fluence of 10^{27} D/m^2 at surface temperatures ranging from 400 K to 950 K. The main focus of this study was put on surface morphology changes as a function of the surface temperature and the evolution of the surface composition, in particular, the W enrichment. Furthermore, the erosion and the D retention were studied. Surface morphology was investigated before and after exposure by scanning electron microscopy (SEM) and confocal laser scanning microscopy (CLSM). Erosion was studied by weight loss and marker technique using focused ion beam (FIB), the D retention by nuclear reaction analysis (NRA), and the W and Ta enrichment by Rutherford backscattering (RBS) and energy dispersive X-ray spectroscopy (EDX).

The surface morphology induced by the plasma exposure varies strongly with surface temperature from needle-like to corral-like structures. The visible lateral length scale of the formed structures is in the range of tens of nanometres to above 1 µm and exhibits two thermally activated regimes below and above ~770 K with activation energies of 0.2 eV and 1.3 eV, respectively. The enrichment of heavy elements at the surface is correlated to the surface morphology. The erosion depends on the accumulated particle fluence and above about 770 K also on the sample temperature. Deuterium is mainly retained in the top 500 nm. Its amount is almost independent of the exposure temperature and is of the order of 10^{18} D/m^2. This small amount corresponds to a fraction of a D monolayer.

Material Transport in JET during the First JET ITER-like-Wall Campaign

Erosion and deposition were studied in the JET divertor and at the JET main chamber inner wall during the first JET ITER-like wall campaign 2011-2012 using marker tiles and long-term samples. An almost complete poloidal divertor section consisting of W-coated carbon-fibre composite tiles 0, 1, 3, 4, 6, 7, 8 and lamellas from the bulk tungsten tile 5 were studied as well as a number of samples from recessed areas at the inner wall cladding between the inner wall guard limiters and from remote divertor areas.

The total amount of material deposited in the divertor decreased by a factor of 4-9 compared to the deposition of carbon during all-carbon JET operation before 2010. Deposits in 2011-2012 consist mainly of beryllium with 5-20 at.% of carbon and oxygen, respectively, and small amounts of Ni, Cr, Fe and W. Deposits are mainly observed on inner divertor tiles 0 and 1, thin deposits are also observed on the rough surfaces of tiles 4 and 6. Erosion is observed on the vertical inner strike point tile 3 and on the bulk tungsten tile 5, where the outer strike point was usually placed and which consequently received the highest particle and power fluxes. The erosion pattern on tile 5 was very complicated and varied in poloidal and toroidal direction on each lamella and between different lamellae due to the shaping of the plasma-exposed lamellae surfaces, shadowing by neighbouring lamellae, and shadowing of lamellae stacks by neighbouring stacks. Shadowed lamellae showed almost no erosion, while the highest erosion was observed on the toroidally curved part of the lamellae surfaces of stack C where the magnetic field lines have the steepest inclination angle. The observed erosion pattern correlates with the strike point distribution.

Transport of beryllium to remote areas of the JET divertor was studied with cavity samples in the inner and outer divertor and below divertor tile 5. Predominantly beryllium films were formed inside the cavities with some additional carbon, the ratio Be/C was >2. These films had high D/(Be+C) ratios of about 0.3. The formation of these films is mainly due to sticking of beryllium-containing particles with low sticking coefficients < 0.5. The observed surface loss probabilities depend on the position in the divertor. The particles responsible for film deposition originated from areas at the divertor strike points.

The observed decrease of material deposition in the divertor is accompanied by a decrease of total deuterium retention inside the JET vessel by a factor of 10-20. The detailed erosion/deposition pattern in the divertor with the ITER-like wall configuration shows rigorous changes compared to the pattern with the all-carbon JET configuration.

Nitrogen Transport in ASDEX Upgrade: Role of Surface Roughness and Transport to the Main Wall

The migration of wall material or seeding impurities plays an important role in the formation of mixed materials, the impurity contamination of the plasma and tritium retention.

The retention of nitrogen in surfaces with varying roughness and the transport of nitrogen from the divertor to the outer mid-plane were studied in experiments at ASDEX Upgrade. To allow for a reliable identification of nitrogen retained during the plasma exposure, 5.3×10^{21} atoms of the tracer isotope ^{15}N were injected into the private flux region of the plasma. On polished W samples exposed to the plasma in the outer divertor, the ^{15}N content peaks to both sides of the strike line with an areal density of 1.5×10^{20} N/m^2 and drops to a value of 1.0×10^{20} N/m^2 at the strike line region. In contrast, the N content of samples with a rougher surface peaks at the strike line and reaches areal densities of 3.0×10^{20} N/m^2.
The dip in the ^{15}N profile at the strike line for the polished samples can be attributed to increased re-erosion and saturation of the ^{15}N retention above and below the strike line. The fact that for rough samples the N retention seems to exceed the commonly accepted value of 1.5×10^{20} N/m^2 for saturation of N retention in W can be understood if co-deposition of N with B is taken into account. A scan of the B profile across the sample reveals the same shape in B deposition as for ^{15}N. On the rough samples these co-deposits are formed in small shadowed regions which are shielded from the incident energetic particle flux responsible for re-erosion. These shadowed regions arise from the roughness of the surface and the oblique incident angles of the ions.
The ^{15}N deposition at limiters in the outer midplane was measured via samples exposed on the AUG-mid-plane manipulator. At a radial location corresponding to the AUG limiter position, the ^{15}N areal density reaches a value of 0.2×10^{20} N/m^2, only a factor of ten smaller than the areal densities in the divertor. This confirms that the main wall contributes significantly to the N storage in the plasma-facing surfaces, as predicted by WallDYN. However, a quantitative comparison to WallDYN simulations employing a model which aims to include the effect of the 3D wall structures in the 2D WallDYN simulations, predicts N areal densities which are by a factor of 4-20 higher than the ones observed in the experiment. Possible reasons for this discrepancy are the toroidal asymmetric main wall geometry, which currently cannot be fully included in the simulations, or an enhanced re-erosion of deposited N.

Tritium Inventory – Understanding and Control

Deuterium Retention in MeV Self-Implanted Tungsten: Influence of Damaging Dose Rate

Hydrogen isotopes retention in future fusion devices with tungsten walls will be largely dominated by trapping due to lattice defects in W created by fast fusion neutrons. To study the effect of lattice damage on H isotopes retention in many cases ions with energies of tens of keV to MeV are used to create displacement damage. Contrary to neutron irradiation, ion-beam irradiation is fast and does not activate the samples. However, it is still unclear to which extent the results gained with ion-beam-damaged material can be transferred to neutron-damaged material. One parameter that was not addressed yet is the vast difference in the damage-creation rate between ion-beam damaging and damage created by fusion neutrons. For studying this issue, recrystallized, polycrystalline tungsten was self-damaged by 20 MeV W ions up to a calculated damage dose in the damage peak of 0.22 dpa. The average damaging dose rate was varied within three orders of magnitude from 5×10^{-3} to 4×10^{-6} dpa/sec, the latter coming close to the damage dose rate expected from fusion neutrons in future devices such as ITER and DEMO. One damaging series was conducted at 295 K and one at 800 K to check for possible effects of defect evolution at elevated temperature. The created damage was afterwards decorated with deuterium until saturation to derive a measure for the defect density that can retain hydrogen isotopes. ^3He nuclear reaction analysis (NRA) was applied to determine the deuterium depth profile and the maximum concentration in the damage peak. For both investigated temperatures no variation in deuterium retention with damage dose rate was found. This observation supports the applicability of high rate self-ion implantation being a valid method to prepare displacement-damaged tungsten as proxy material for retention studies with neutron-damaged tungsten.

Diffusion-Trapping Modelling of Hydrogen Recycling in Tungsten under ELM-like Heat Loads

The recycling of D ions impinging onto a W divertor surface is a key input parameter into the power and momentum balance at the target boundary during SOL modelling. It is described by the ratio R of the flux of recombining D_2 molecules to the non-reflected incident ion flux. In steady-state plasmas where the surface is in equilibrium with the incident flux, R equals one due to particle conservation. However, during transient events such as edge localized modes (ELMs) the evolution of R with time is not straightforward to predict. Therefore, detailed diffusion-trapping calculations were performed taking into account the variations in power influx and particle energy during an ELM. They showed that in contrast to the naive expectation, that the ELM would deplete the surface and subsequently lead to ‘pumping’ ($R\ll1$) of the incident flux by the empty surface, $R\approx1$ or even $R>1$ occurs. The reason is for this evolution during and after the ELM is that the surface equilibrates almost instantaneously with the incident flux by changing the surface solute D concentration gradient to counter balance the increased influx with an increase in the out-diffusion flux. This equilibration with the incident flux leads to values of $R\sim0.999$ during- and 1.001 after the ELM. To understand this evolution better an analytical model for R was developed which allows qualitatively understanding its evolution as calculated by the diffusion-trapping code.

These calculations were performed with approximately "ELM like" loads: Gauss shaped excursions in power & particle flux with 1 Hz repetition frequency and no spatial resolution. These calculations were recently extended to include the incident power and particle flux distribution from real ELM-averaged data with poloidal resolution. These calculations showed the same qualitative result: R~1 holds throughout the entire ELM footprint.

Materials and Components

Characterisation and Optimisation of Self-Passivating Tungsten Alloys

Tungsten is presently the main candidate material for the first-wall armour of future fusion reactors. However, if a loss of coolant accident with simultaneous air ingress into the vacuum vessel occurs, the temperature of the in-vessel components would exceed 1000 °C, leading to the undesirable formation of volatile and radioactive tungsten oxides. A way to prevent this serious safety issue is the addition of oxide-forming alloying elements to pure tungsten which, in presence of oxygen at high temperatures, promote the development of a self-passivating oxide layer which protects tungsten against further oxidation.

Recently, bulk tungsten alloys of the W-Cr-Y system with different concentrations of the alloying elements were studied in collaboration with CEIT (San Sebastian, Spain) in order to establish their optimum composition for lowest possible oxidation rate and the best self-passivating behaviour together with acceptable thermal and mechanical properties. The materials were manufactured by mechanical alloying and subsequent densification by hot isostatic pressing at CEIT and (partly) thermal treatment at 1550 °C. Microstructural investigations of the bulk material and the thin oxide layer developed were investigated. Without temperature treatment, the W-Cr-Y alloys exhibit an ultrafine grained microstructure with an average grain size around 100 nm, whereas after annealing the grains were in the µm range. The oxidation tests were performed under isothermal (800 °C and 1000 °C, constant oxygen atmosphere) and accident-like conditions (temperature ramps up to 1000 °C, switching of oxygen influx). In both kinds of experiments the W-10Cr-0.5Y alloy exhibits lower oxidation rates than W-12Cr-0.5Y and compared to all previously investigated alloys the Y alloys exhibit a much better oxidation behaviour. The annealing seems to further improve oxidation behaviour, possibly because it suppresses the development of crater like structures on the oxidised surfaces. Four different samples with self-passivating alloys were subjected to high heat flux tests (see also section on GLADIS) with a power density of 3.5 MW/m² using actively cooled samples, reaching surface temperatures of 860 – 960 °C. The applied power density was chosen to be about a factor of 2 higher than the maximum power load expected at the blanket first wall in order to provide some safety margin. Only on a heat treated (WCr10Ti2) sample cracking was observed, whereas no macroscopic damage appeared on the others.

Tungsten Fibre-reinforced Tungsten – The Way to a Mock-up for Divertor Applications

For the next step fusion reactor tungsten is foreseen as plasma-facing material in order to minimize erosion and hydrogen retention and to allow operation at elevated temperature and high heat loads. A major caveat of tungsten is its brittleness at low temperatures and after neutron irradiation. Tungsten fibre-reinforced composites (W_f/W) can overcome this intrinsic brittleness thus allowing its use as a structural as well as an armour material. During recent years small W_f/W samples have been produced and characterized providing the proof of principle for its increased toughness compared to bulk tungsten. The project now tries to provide the proof of concept for the W composite by developing further the production routes towards the capability to produce complete mock-ups of plasma-facing components for thermo-mechanical characterisation and testing. One concept is the use of chemical vapour deposition where W fibre layers are positioned and ingrown into the W-matrix one after the other until the final sample thickness is reached. Textile techniques (weaving, braiding) are utilized to optimise the tungsten wire positioning and the preform production. The sample production is accompanied by modelling which reveals that dense bulk W_f/W can be produced by the use of a convectional flow of the process gas WF_6 and a directed temperature profile during the infiltration process. In parallel it was shown that the use of potassium doped tungsten fibres (as used in the light bulb production) could increase the allowed fabrication temperature and broaden the operation temperature window of W_f/W significantly.

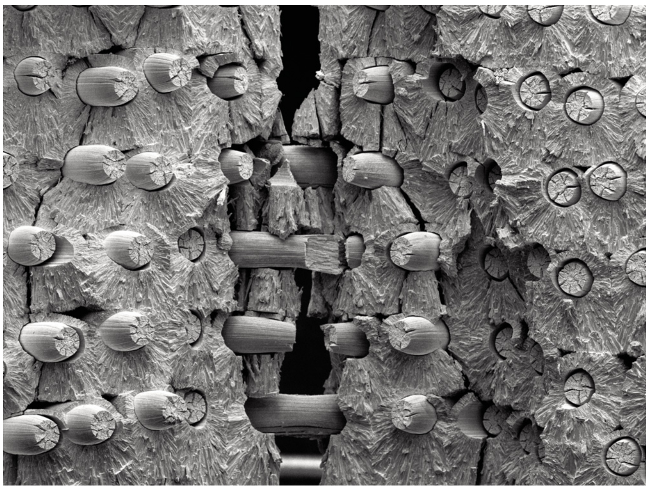

Figure 1: Cross section of W_f /W composite after a Charpy impact test: The tungsten fibres show ductile deformation and pull-out leading to increased toughness of the composite.

Charpy impact tests on samples produced with the new technique showed increased fracture energy mainly due to the ductile deformation of the tungsten fibres (see figure 1). Cyclic bending tests reveal that the extrinsic toughening mechanisms counteracting the crack growth are active and stable. FEM simulations showed that the use of W_f/W can mitigate problems of deep cracking occurring typically in cyclic high heat flux loading (see below).

Crack Growth in Tungsten Monoblocks

The HHF qualification tests conducted on the ITER divertor target prototypes showed that the tungsten monoblock armor suffered from deep cracking due to fatigue, when the applied HHF load approaches 20 MW/m^2. A rigorous theoretical interpretation of the observed cracking features has been conducted by means of finite element analysis. A twofold computational approach was employed: 1) to simulate the plastic fatigue of the armor surface for assessing the lifetime to crack initiation and 2) to compute the crack tip load for estimating the driving force of crack growth. The quantitative predictions delivered in this study agree well with the observed findings offering insight into the mechanisms of deep cracking. Cracks can be initiated on the surface of tungsten armor, if the applied HHF load is about 15 MW/m^2 or higher. At such loads the tungsten armor begins to be partially recrystallized from the surface layer. The major features predicted at 20 MW/m^2 are as follows: 1) The most probable cause of cracking is low cycle fatigue (LCF) damage followed by crack formation due to tensile stress developing in the surface region during cooling stage. 2) LCF damage is promoted by the low yield stress of recrystallized tungsten during HHF heating. Recrystallized tungsten has a much lower LCF life compared to cold-worked tungsten. 3) Once a crack is created on the armor surface, it can grow during the cooling stage where strong tensile stress develops in the upper part of the armor. Crack growth can be continued during the HHF heating stage where strong tensile stress develops in the copper interfacial region of the tungsten armor block and the cooling channel.

High Heat Flux Facility GLADIS

The high heat flux (HHF) test facility GLADIS is equipped with two independently controlled ion sources of up to 1 MW power each. GLADIS offers unique capabilities for the investigation of plasma-facing materials and components. High thermal loads can be generated by H, He or mixed H/He neutral beams. As facility improvement, the installation of a pressurized hot water cooling circuit was completed in 2016. The cooling water temperature up to 230 °C at 4 MPa covers ITER and DEMO divertor cooling parameters.
The main activities were dedicated to the investigation of

- W7-X divertor components and diagnostic development (real-time infrared monitoring),
- W heavy metals (W/Ni-Fe and W/Ni-Cu) as alternative divertor target material in ASDEX Upgrade,
- newly developed materials, e.g. W/Cu compounds and self-passivating W alloys,
- erosion & hydrogen retention of actively cooled W,
- actively water-cooled W components developed in the framework of the European DEMO activities and ITER divertor development.

Plasma-facing components (PFCs) with tungsten monoblock geometry are employed at the vertical targets of the ITER divertor, where the stationary heat flux is highest. The thermal performance and lifetime of the PFCs rely on the quality of the W monoblock-heat sink joints. Only HHF loading can generate thermo-mechanical stress in the component similar to the expected operating conditions in a fusion device. Other non-destructive examinations, as performed by the manufacturer, do not consider this complex thermo-mechanical behaviour during heat loading. Therefore, the investigation of W monoblock PFCs in GLADIS was focussed on the development of a quality assessment method based on a statistical approach originally developed for the HHF assessment of the W7-X divertor manufacturing. This approach evaluates the thermal response of the delivered components compared to the behaviour of successfully tested prototypes, allowing an assessment of industrially manufactured PFCs equipped with ten to hundred thousand of W monoblocks with a reasonable HHF test effort.
It is expected that the adaption of the W7-X approach should be possible, because all analysis and image processing tools are available. However, the following issues must be solved in advance:

- Due to the monoblock geometry, the surface temperature response of debonding is not well defined, in contrast to the flat tile design.
- The precise detection of local temperature differences of about 20 °C at 1000 °C surface temperature is necessary.
- Development of suitable quality criteria taking into account the monoblock design and the corresponding specification limits.

The main challenge for reliable temperature measurements in the mid wavelength IR (~3^{-10} µm) is the variation of the emissivity of the individual W blocks, which depends strongly on the surface temperature, the number of heat cycles and the machining quality as illustrated in figure 2. A correction routine based on simultaneous monochromatically corrected two-colour pyrometer measurements was developed and applied on the IR raw data. One hundred cycles at 10 MW/m^2 were performed on eight components which were manufactured in EU, China and Japan for ITER, WEST and the DEMO divertor development. As a first result, we obtained a Gaussian distribution of the measured ΔT as an indication of stable manufacturing processes.

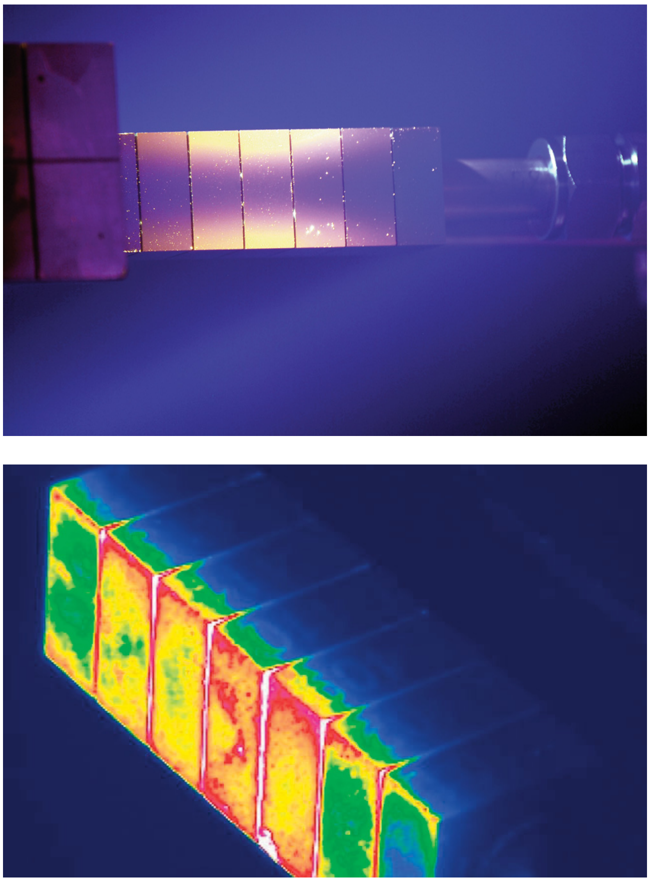

Figure 2: The upper image gives an impression of the surface temperature distribution of a W monoblock component during HHF loading at 10 MW/m². Due to the geometry of the W blocks, the outer edges are slightly hotter than the central temperature of 925 °C. The lower IR image shows very clearly the variation of the local emissivity resulting in an inhomogeneous temperature signal.

Integration of and Collaboration in the EU Programs

Within the EUROfusion work programme many members of the project are involved in the scientific exploitation of the European tokamaks under the Taskforces WPJET1 and WPMST1, but the project's main activities under EUROfusion are centred within the work packages WPPFC, WPMAT, WPDIV and WPJET2.

Work Package Plasma Facing Materials (WPPFC)

The work within WPPFC comprises seven subprojects dealing with subjects centred around investigations on plasma-material interaction mostly in linear and laboratory devices. IPP is strongly involved in five of the subprojects and provides the subproject leaders for 'Erosion, deposition and mixing' and 'Fuel retention, fuel removal and damage'.

Work Package Materials (WPMAT)

Within WPMAT materials for a Demonstration Fusion Power Plant are developed. The work of IPP concentrates on the development of high heat flux materials namely tungsten fibre-reinforced tungsten, copper composites as well as the self-passivating tungsten alloys as an armour material for the main chamber plasma-facing components.

Work Package Divertor (WPDIV)

The scope of the Divertor Project (WPDIV) covers all elements of conceptual design for the whole divertor system of the first DEMO reactor. IPP provides the project leader of WPDIV and is responsible for the design and construction of two divertor component mock-ups. Further, the project provides the thermo-mechanical analysis of all European mock-ups and their high heat flux testing in GLADIS.

Work Package JET2 (WPJET2)

The work package JET2 (WPJET2) aims at the exploitation of the JET ITER-Like Wall (ILW) in view of the erosion/deposition pattern and fuel inventory of the beryllium and tungsten plasma-facing components. IPP is responsible for the nuclear reaction analysis of specific tungsten components.

Scientific Staff

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Plasma Theory

Theoretical Plasma Physics

Heads: Prof. Dr. Per Helander, Prof. Dr. Emanuele Poli, Prof. Dr. Eric Sonnendrücker

The project "Theoretical Plasma Physics" is devoted to first-principle based model development with emphasis on magnetic confinement. It combines the efforts of the divisions Tokamak Physics, Stellarator Theory and Numerical Methods in Plasma Physics and of the HLST Core Team of the EUROfusion work package WPISA. It is also a major partner in the Max Planck Princeton Center for Plasma Physics.

Tokamak Physics Division

Head: Prof. Dr. Emanuele Poli

Tokamak Edge Physics

Support of the SOLPS edge plasma simulation code continued, including code developments and interactions with code users. In the past much of the development in SOLPS has been spent in improving the physics treated by the code. This increase in fidelity usually came at the cost of slowing down the code. In order to explore operations on possible future devices, the possibility of stripping out some of the physics options has been investigated. This is done (1) by using a fluid neutral model rather than the more expensive (and exact) kinetic model, (2) by using coarser grids that still reproduce the trends found on finer grids, and (3) by using aggressive charge state bundling for impurities so that running with, for example Xe, is as expensive as running He rather than ~18 times as expensive. This has enabled the simulation of D+T+He+Ne+Ar+Kr plasmas for ITER size machines with DEMO type powers in days rather than the years it would have taken for the full model. In work within the EUROfusion Work Package on Code Development for Integrated Modelling, SOLPS has been coupled to the European Transport Simulator core model to allow for coupled core-edge calculations.

The influence of edge physics (in the SOL and divertor) on the H-mode power threshold was studied. EDGE2D-EIRENE code simulations were made for two magnetic configurations in the divertor, with the outer strike point on the horizontal (HT) or the vertical target (VT). In the experiment, the H-mode power threshold is ~2 times lower in the HT configuration, all other main plasma parameters and profiles being identical in the two configurations. The simulations produced very different profiles of radial electric field E_r in the SOL, with a pronounced positive E_r spike in the HT configuration. It was hypothesized that such a spike and a larger E×B shear across the separatrix associated with it could explain the easier transition to the H-mode (at lower input power) in the HT configuration. The E_r spike was related to ballistic effects of neutrals reflection from the outer strike point, with neutrals tending to be ionized along the power conduction channel in the VT configuration. This leads to higher plasma density and lower electron temperature at the strike point in this configuration and, via the mechanism of the target Debye sheath, produces different E_r profiles in the main SOL.

The KInetic code for Plasma Periphery (KIPP) was further developed by including effects of curvilinear toroidal geometry, allowing it to analyse kinetic effects of plasma transport in realistic magnetic configurations. The 1D spatial grid acquired a variable cross-section, with constant magnetic flux across its boundaries (faces), and toroidal effects (e.g. 'mirror force') were also included. In test cases with high collisionality it was demonstrated that additional terms related to a curvilinear toroidal geometry cancel each other out, as expected for highly collisional near-Maxwellian electron distribution functions. At high collisionality KIPP versions for the Cartesian and curvilinear toroidal geometry produced nearly identical results, indicating the correctness of the implementation of the new terms.

The code GRILLIX, based on the flux-coordinate independent approach, is able to treat the complex geometry of edge/SOL in diverted devices, i.e. separatrix with X-point(s). The behaviour of turbulent structures around the separatrix was studied in realistic geometries. In order to elucidate the effect of X-point(s) on turbulent structures, simulations in a purely closed field line domain, in a domain with a single X-point and in a double-null configuration were carried out. Turbulence is mainly driven around the outer midplane region, where magnetic curvature is unfavourable, and in a closed field line domain the fluctuations are then mediated to the high field side via parallel dynamics. Near X-point(s) this parallel dynamics is disabled due to the strong local magnetic shear separating the outboard side from the stabilizing inboard side. Therefore, especially in a double-null configuration, fluctuations are strongly supressed on the high field side. Besides continuous improvement of the numerics, recent and current efforts have concentrated on the implementation of a drift-reduced-full-f fluid model into GRILLIX. With the relaxation of the Boussinesq approximation a nonlinear elliptic problem has to be solved in each time-step, for which an efficient geometric multigrid solver was developed. The code was extended by a parallel transport model and inclusion of electron thermal fluctuations is ongoing.

MHD Theory

Equilibrium Calculation and Stability Analysis

The 3D resistive linear stability code CASTOR3D is under development, based on the combination of the codes CASTOR_3DW and STARWALL. Numerous modifications and extensions of both code parts led to a synergistic effect, the number of possible applications of CASTOR3D exceeds easily the capabilities of both of them. The code has a number of significant advantages. It allows us to (i) choose between various kinds of flux coordinates, (ii) perform stability studies for 3D, ideal and resistive tokamak equilibria, (iii) take simultaneously into account plasma inertia and resistive walls, (iv) investigate vertical instabilities, and (v) deal with coils and multiply-connected wall structures.

Benchmark calculations show an excellent agreement between the CASTOR3D results and the results obtained with CAS3DN, STARWALL, and the previous CASTOR version. The latter is restricted to axisymmetric equilibria and straight field line coordinates. Several of the CASTOR3D computations were performed for NEMEC, Boozer, and 2D straight field line coordinates. All the results obtained using these coordinates agree very well. They show that an appropriate choice of the coordinates may noticeably reduce the number of required poloidal and toroidal harmonics, thus reducing computing time and memory.
The CLISTE code was extended as follows: (i) Polarimetry data from the interferometer system was added as an additional constraint and good agreement with validated diagnostic data was obtained. (ii) The Grad-Shafranov equation may be solved using an alternative set of source functions based on a symmetric/antisymmetric partition of the current density source terms. (iii) CLISTE has been extended to solve multi-time-point data by including a linear time dependence on all fitted parameters.

Nonlinear MHD

In nonlinear MHD simulations with JOREK based on ASDEX Upgrade H-Mode equilibria saturated edge modes have been observed, which lead to moderate confinement loss and may be responsible for limiting the pedestal pressure gradients in the experiment. The properties of these modes depend on the interplay of stabilizing and destabilizing effects. Cases are observed where a stationary saturated mode similar to the edge harmonic oscillation (EHO) in QH-Mode plasmas is developed (whether this occurs at comparable parameters needs to be studied), while in other cases precursor-like structures exist for several milliseconds before a small edge localized mode (ELM) crash occurs (figure 1).

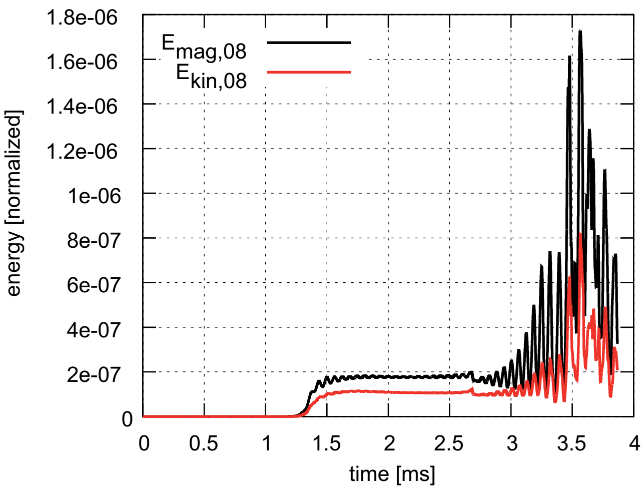

Figure 1: Simulation of the edge MHD activity for ASDEX Upgrade discharge 31128. A precursor-like structure is observed for several milliseconds followed by a small ELM crash.

The dynamics of ELM filaments shows qualitative agreement with experimental measurements. Heat flux patterns and the heat flux asymmetry between inner and outer divertor target were studied for large ELM crashes in normal and reversed field configurations including all relevant plasma flows.
The penetration of resonant magnetic perturbation (RMP) fields into ASDEX Upgrade H-Mode plasmas was studied for different phases between upper and lower B-coil currents with nonlinear MHD simulations showing good agreement with other modelling and the experiment, e.g., the field line excursion at the midplane and X-point for different coil current phases. The strongest ELM mitigation observed in experiments corresponds to the largest plasma resonant and edge kink responses to RMPs found in modelling. Moderate density pump-out was observed in the simulations, but until now smaller than found in the experiment. The implementation of the boundary conditions for RMPs is currently being improved for the nonlinear simulations by using JOREK-STARWALL to obtain a better description of field amplification in the plasma edge and scrape-off layer. Nonlinear studies of the interaction of RMP fields with ELMs have been started aiming at identifying the mechanisms of ELM suppression and mitigation. Current work aims at comparing quantitatively the heat flux patterns induced by RMP mitigated ELMs on divertor with infrared measurements. ELM triggering by pellet or vertical kick is studied in ASDEX Upgrade geometry in close collaboration with external researchers.
Disruptions triggered by deuterium massive gas injection into an ASDEX Upgrade H-mode plasma have been studied with nonlinear JOREK simulations, in particular the dependency on key physics parameters. Cases are observed in which the plasma survives the gas injection, while it undergoes a sudden thermal quench in other cases, e.g. at higher resistivity, lower rotation, or more gas injected. Experimental parameters can be matched fairly closely in the simulations.
Based on the “four field equations”, the numerical code TM1.f has been upgraded to include the toroidal coupling of modes with the same toroidal mode number but different poloidal mode numbers, in addition to nonlinear two-fluid effects. Straight field line coordinates are used, and the cold ion assumption is made. Benchmarks with other numerical codes have been carried out successfully for the case of a single fluid without rotation, and comparison with other 3D MHD instability codes is going on.
Using two-fluid equations in cylinder geometry, calculations of the nonlinear growth of internal kink modes have been performed with the following findings. (a) The simulated fast sawtooth crash time agrees with experimental one. For single fluid simulations, the formation of plasmoids has been found to accelerate magnetic reconnection, as described in previous work for the case of astrophysical plasmas. (b) A shear plasma rotation inside the q=1 surface, driven by the electromagnetic torque during mode growth, is found.

The parallel electric field is increased by three orders of magnitude during the sawtooth crash in the island's x-point region, possibly generating super-thermal electrons in a fusion reactor. (c) Calculations using ASDEX Upgrade parameters reveal a possible role of m=2 perturbations in affecting the m=1 mode growth, which can result in a slow growth phase of the internal kink mode in some cases, providing a possible explanation for the corresponding experimental observation.

A method to stabilize neoclassical tearing modes by using radio frequency current drive and static resonant magnetic perturbations is investigated. The stabilization is found to be enhanced if the RMP phase has a half period difference from that of the RF wave deposition along the helical angle. The required RF current for NTM stabilization is reduced by about one third if an appropriate RMP amplitude is applied.

A method for accelerating magnetic island rotation by modulated RMP is investigated. Nonlinear modelling shows that the island can be completely or substantially suppressed. The mode frequency is significantly increased in the second case, suggesting a possible method for avoiding mode locking.

Energetic Particle Physics

Motivated by theoretical predictions, in off-axis neutral beam injection experiments at ASDEX Upgrade, a low background plasma temperature (both ions and electrons) despite intense beam injection could be established by letting tungsten impurities accumulate in the core. This leads to a large ratio of beam pressure to background pressure (≥1) similar to values in present-day DEMO studies. In these scenarios, the strong instability drive causes the behaviour of energetic-particle-driven modes (EPM) to be very different from well-documented cases with a lower drive. Strong toroidal Alfvén eigenmode bursts occur, together with chirping energetic-particle driven geodesic acoustic modes. The nonlinear mode structure evolution could be measured and phase space coupling effects between the modes could be documented. A linear gyro-kinetic analysis (LIGKA) confirms the mode onset thresholds for different heating powers.

The global version of LIGKA contains all physics elements for linear stability studies but is too slow and too expensive to be used as a module in a transport code or for comprehensive sensitivity studies. The local analytical LIGKA version is suitable for this task but does not include FLR and FOW effects that are crucial for EP physics. A model that uses a 2nd order Taylor expansion of the radial excursion of the particle orbits has been developed such that in the appropriate limit the dispersion relation derived by Zonca is exactly reproduced. This will give LIGKA the flexibility to work either with analytical estimates (now including FLR and FOW) effects, with semi-analytical expressions (in case of non-Maxwellian distribution functions) or fully numerical orbit integrals (HAGIS). The new model can be used to obtain a fast and accurate linear stability diagram for all toroidal mode numbers of interest, including EPM thresholds. This is a very important building block for a gyrokinetic quasi-linear EP transport model.

The Aflvénic fluctuations excited by anisotropic EPs at low magnetic shear have been investigated with XHMGC. Saturation mechanisms due to radial decoupling or resonance detuning are studied for circulating fast ions. These two processes cause the observed differences of saturation amplitude and its scaling with the mode growth rate, as illustrated by a simple nonlinear pendulum model. Which regime occurs can be predicted on the basis of linear dynamics: the radial profile of the fast-ion resonance frequency, the mode structure, and the considered toroidal number. We also analysed EPM saturation and the corresponding frequency chirping. Phase locking has been proposed, within the fishbone paradigm, to describe such chirping: the resonance with linearly resonant particles is maintained through a continuous variation of the mode frequency. We show that an additional scenario is possible: mode radial localization and frequency appear to be locked to the shear Alfvén continuum; once the linear resonance population has exhausted its driving capability (by local flattening of the phase-space distribution function) the mode is shifted to non-exhausted regions of the phase space. The effect is a succession of resonant excitations from different phase-space regions (each characterized by its own nonlinear evolution time), rather than mode adjustment to the evolution of the linearly-resonant particles.

Collisionless Vlasov-Poisson simulations of the electron bump-on-tail system with multiple modes have been performed with both Eulerian and particle based tools. A transport threshold has been observed above which phase space redistribution is enhanced as a result of the overlap of phase space structures. The ability to track particle trajectories can be a powerful tool for mapping the phase space position of a mode, which allows us to confirm the claim that the threshold of enhanced transport corresponds to the intersection in phase space of different modes. Eulerian methods suffer spurious diffusion when the grid cells are no longer sufficiently small to resolve the scales of the mode. In contrast, we see that particle methods tend to under-predict the transport as the number of markers is decreased. We postulate that if particle methods are missing the critical trajectories through which the modes overlap, then the mechanism of the interaction is missed, and the simulation lacks the increased transport.

Kinetic Theory and Wave Physics

The verification and validation process for the wave-kinetic-equation solver WKBeam, developed jointly with the division Numerical Method for Plasma Physics, has entered its final phase. The code calculates propagation and absorption of EC beams retaining the statistically averaged effect of turbulent density fluctuations, which lead to a broadening of the beam width and thus loss of EC power localization.

The underlying physical model relies on a perturbative treatment (Born approximation) of the scattering term. The validity of this approach under ITER heating and current-drive conditions has been assessed through a thorough comparison with the full wave solver IPF-FDMC (University of Stuttgart), for which similar dimensionless parameters have been employed. Noticeable differences between both solvers can be seen only starting from unrealistically high fluctuation levels. A reference "fluctuation amplitude profile", based on results of present and past tokamak experiments has been established, according to which the relative density fluctuations are increased from a level below few percent to 20-40% at the outer edge of the H-mode pedestal, and can further increase in the scrape-off layer. These investigations constitute the basis for the just-started extensive analysis of the problem under ITER conditions, where the beam spreading is expected to have a significant impact on the ECCD profile (see ITER chapter).

The analysis of the impact of ICRF heating on in-out asymmetry of high-Z impurity species has been extended to address the consequences of toroidal effects. This has been accomplished with Dendy's model for the distribution function of the heated species. Because of its effect of increasing the fraction of trapped particles with their banana tips on the IC resonance layer, ICRF heating produces a local maximum of the heated species around the resonance layer. As a consequence, the impact of ICRF heating depends substantially on the location of the wave absorption, and is strongly reduced when the resonance location is moved from the low-field side to the high-field side. This is in qualitative agreement with experimental findings in Alcator C-Mod, ASDEX-Upgrade and JET.

The first phase of the benchmark activity on full-wave ICRF codes in the ITM European framework has been completed and documented. The benchmark was extended to non-European codes such as AORSA of Oak Ridge. An important spin off of this activity is the creation of a set of regression testing cases useful for future code developments and import procedures in the ITM framework. The new three-ion ICRF heating scenarios proposed by Ye. Kazakov for ITER and W7-X have been studied with the full-wave code TORIC. Investigations of ICRF heating efficiency for the D-T campaign on JET have been also done.

The effect of islands and of fast toroidal rotation on the neoclassical transport of W ions in a tokamak plasma was studied with guiding centre particle simulations assuming the trace limit. The transport is reduced in the island due to the flattening of the main ion profiles. Strong toroidal rotation increases the local radial particle flux everywhere except at the midplane. If the density is localised poloidally and shifted away from the midplane, the surface averaged radial flux increases strongly. This result implies that a narrow localised impurity density has to be centred near the mid-plane.

Plasma acceleration by the j×B force due to a current between electrodes was studied with PIC simulations (resolving the gyro motion) in a slab model with two rod electrodes floating in the plasma with a potential difference U between them, the cathode is emitting electrons. There are two regimes characterised by the values of ρ_{iU} and ρ_{eU} [$\rho_{aU} = (2eUm_a)^{1/2}/eB$]. For $\rho_{iU} > d$ (distance between electrodes) a strong ion current can flow across the magnetic field lines provided a particle source is delivering the electrons to the anode and the ions that carry the cross-field current. Maximum spin-up of the plasma is obtained for $\rho_{iU} \approx d$; electron emission at the cathode has almost no effect on the momentum transfer. For $\rho_{eU} > d$ the electrons emitted by the cathode can flow to the anode carrying a large cross-field current. The momentum transfer to the electrons is determined by the electron emission at the cathode.

Transport Analysis

The implementation of first order neoclassical corrections, as computed by the drift kinetic code NEO, in the background distribution function of the gyrokinetic code GKW has been completed. Coupled NEO and GKW calculations were extensively performed and applied to compare the related predictions of toroidal momentum residual stress with observations of intrinsic rotation in ASDEX Upgrade OH plasmas. It is found that the inclusion of neoclassical corrections in the background of the gyrokinetic description produces residual stresses which are non-negligible and comparable to other known symmetry breaking mechanisms, but still too small as compared to the observed values of intrinsic toroidal rotation velocity gradients. An analysis of the uncertainties confirms that this mechanism alone cannot explain the observations. A companion work has clarified the impact of these neoclassical corrections on turbulent impurity and momentum transport by means of the analytical derivation of a fluid model combined with numerical gyrokinetic calculations. This study shows that neoclassical corrections can also have non-negligible impact on impurity transport, particularly for heavy impurities at high collisionality of the bulk plasma. The intrinsic rotation database of ASDEX Upgrade Ohmic L-mode plasmas has been also analysed by means of a realistic momentum transport equation introduced in the ASTRA code. The role of impurity-deuterium differential rotation in the interpretation of the experiments and in the comparison with theory has been clarified.

The physics behind the transition from linear (LOC) to saturated (SOC) OH confinement in ASDEX Upgrade was investigated by means of simulations with the TGLF transport model coupled to ASTRA. The modelling reproduces the observed transition from LOC to SOC energy confinement time. The increasing thermal exchange from electrons to ions with increasing density and the stronger ion stiffness produced by ITG turbulence are the main elements producing

the LOC-SOC transition, accompanied by TEM stabilization due to increasing collisionality. This impacts the behaviour of the density profile, which is well reproduced by TGLF. The qualitative behaviour of the global confinement is well reproduced, but the logarithmic ion temperature gradients are over-predicted by the TGLF modelling.
A gyrokinetic study was dedicated to clarify the impact of the electron to ion heating ratio on the heavy impurity transport. It is shown that at constant total heat flux, the impurity diffusivity is a non-monotonic function of the heat flux ratio and can vary by one order of magnitude, reaching its maximum when the electron heat flux is comparable or slightly exceeds the ion heat flux. In these conditions subdominant modes can also impact the turbulent convective components of the heavy impurity transport.

Turbulence Theory

Global Alfvén dynamics with and without fast particle effects have been studied with both types of models, gyrokinetic and gyrofluid, with special emphasis on the ability to capture continuum damping of both Alfvén and geodesic acoustic oscillations. A main point has been the nonlinear interaction between the three channels of dynamics represented by Alfvén and geodesic responses and drive and saturation of energetic-particle drive effects via velocity space nonlinearity.
The combined effect of phase mixing and Landau damping on geodesic acoustic oscillations was investigated by means of analytical theory and linear gyrokinetic simulations with the gyrokinetic PIC model ORB5. In particular, the cascade of the radial wavenumber (successive increase through interaction with the geometry) in time was investigated numerically, with a specific initial value model also built for dedicated study of the phase-mixing. This cascade in wavenumber has been shown to be responsible for increasing the effective collisionless damping of GAMs by up to an order of magnitude for realistic tokamak conditions. Further work will be done, to quantify how this mechanism can play a role in the nonlinear interaction of GAMs and turbulence, and the suppression of GAM oscillations in the regions characterized by a strongly non-uniform temperature profile such as in the H and I modes.
A numerical investigation of the linear and nonlinear dynamics of Alfvén instabilities driven by energetic particles was performed with ORB5. The linear phases have been studied within the international ITPA benchmark effort on toroidicity-induced Alfvén eigenmodes (TAEs), which involves both gyrofluid and gyrokinetic models. The radial structure of the eigenmodes has been studied, as well as the relationship of the mode frequency to the gaps in the continuum spectrum. The nonlinear phases involve both wave-particle and wave-wave interaction. In the case of wave-particle interaction, the nonlinear modification of the frequency and radial structure has been quantitatively described for TAEs in the ITPA equilibrium, and the saturation levels have been compared with other gyrokinetic and hybrid gyrokinetic/MHD codes. A transition from quadratic to linear scaling of the saturation amplitude with linear growth rate is found in ORB5 simulations (figure 2) in agreement with the theoretical analysis (see Energetic Particle Physics). Study of wave-wave interaction yielded excitation of zonal structures which were found to dominate the Alfvén mode saturation in simplified equilibria (with flat q profile).

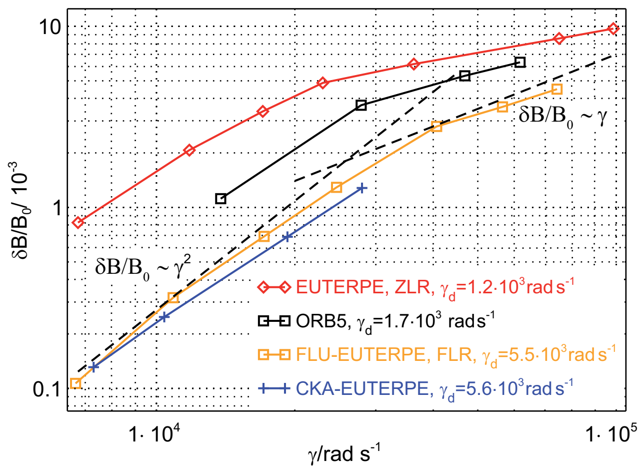

Figure 2: Linear growth rate γ plotted against saturated perturbation amplitude $\delta B/B_0$ in the nonlinear m=10; 11, n=-6 ITPA TAE case, for perturbative hybrid model CKA-EUTERPE and non-perturbative hybrid model FLU-EUTERPE with FLR effects, and fully gyrokinetic codes EUTERPE and ORB5 without FLR effects.

The self-consistent generation of the tokamak bootstrap current due to finite density and temperature gradients was studied with the total-f continuum gyrokinetic model FEFI. The successive development of the Pfirsch-Schlüter current, due to Alfvénic force balance along the magnetic field lines, and then the bootstrap current on the ion collision time scale, were elucidated by varying the time scale separation. This also pertains to the transport time. For conventional mid-size tokamaks the ion neoclassical transport time overlaps the collisional relaxation sufficiently that these two processes are not separate in the pedestal. For ITER cases, however, the ion neoclassical transport time is long enough that ion temperature relaxation during a run is negligible. This may have a measurable effect on hysteresis in the various relaxation processes during ITER ELM cycles, making them operationally different from those on present-day tokamaks.
A version of the derivation of the gyrokinetic Lagrangian was done, which uses the Littlejohn gauge transformation method rather than Lie transforms, which dispenses with any splitting between equilibrium and dynamical space scales for dynamical field variables. The shear-Alfvén electromagnetic Lagrangian used in the momentum conservation proofs is recovered.

Correspondence between this model and the more general finite-beta electromagnetic model of Hahm, Lee, Brizard was shown for low E×B Mach number and conventional tokamak ordering with respect to both beta and aspect ratio. Hence, gyrokinetic theory as we know it is on strong foundation for the pedestal region of present-day conventional tokamaks and ITER.

Turbulence in Laboratory Plasmas

The main goal of our research efforts is to develop a better understanding of the nature of plasma turbulence. In close collaboration with national and international partners, we work on extensions of our tool, the massively parallel simulation code GENE, and its applications to existing experiments and predictions for future devices. Over the last few years, gyrokinetic turbulence simulation has matured to a point where the nonlinear simulation results can be compared directly with heat flux levels from tokamak experiments and also with density and temperature fluctuation levels, allowing for a constructive dialogue between first-principles theory and measurements. It is necessary to properly assess and represent the experimental diagnostics and implement corresponding synthetic diagnostics in the numerical framework. Based on previous work successfully reproducing Correlation Electron Cyclotron Emission and assessing Beam Emission Spectroscopy measurements in the DIII-D tokamak, we were able to provide theoretical support for the development of a CECE system at ASDEX-Upgrade. Further studies were dedicated to comparisons with wavenumber resolved Doppler reflectometry data, where, e.g., spectral breaks and scaling exponents have been found in good agreement with experiment. Such joint studies are particularly important as they establish a sound basis for further investigations of fundamental turbulence aspects like the development of turbulence cascades and the interplay of various spatial and temporal scales.

Applying gyrokinetic codes to realistic scenarios, electromagnetic (beta) effects have been identified as an important simulation component in many situations during the last years. For instance, in the inner core where the magnetic shear is low they tend to be a crucial ingredient. Furthermore, the presence of a sufficiently large fast-ion population in strongly electromagnetic plasmas has been shown to allow access to steep ion temperature profiles. With the overall goal to reveal necessary physics effects to be included in simplified models for scenario development, this previous work has been further extended. E.g., a JET-ILW power scan has been studied, allowing for a direct comparison of 6 MW and 12 MW NBI heated discharges. We show that increased beta plays a stabilizing role and allows the ion temperature gradient to be larger at high power. In a sensitivity study on the safety factor q, we found that the impact of fast ions is negligible at low q also in the 12 MW heated case. This is because beta-stabilization becomes weaker as the KBM threshold is shifted further away from the experimental operation point due to increased q. Moreover, cross-dependencies between beta and the electron-ion temperature ratio have been identified. For ASDEX Upgrade a gyrokinetic analysis of a non-inductive discharge with a large fast ion fraction has been performed. For this discharge the employed quasi-linear codes failed to reproduce the steep ion temperature profile. Linear gyrokinetic simulations indicate that NBI fast ions can be made responsible. For current European DEMO scenarios first gyrokinetic studies with particular emphasis on the possibly beneficial influence of active fast ion species were performed. Here, the existing predictions based on quasilinear modelling apparently require improved models for the saturation rules even in the absence of fast ions and electromagnetic effects.

A crucial question in the context of fast ion enhanced electromagnetic stabilization is the impact of the non-fluctuating component of the distribution function. We assume in our numerical framework Maxwellian distributions on each flux surface. Fast ions such as fusion alpha particles or Ion Cyclotron Resonance heated or Neutral Beam Injected particles may require more sophisticated descriptions based on dedicated numerical tools or approximated by analytic slowing-down or bi-Maxwellian descriptions. Our code is being updated to consider such background distributions and has demonstrated already good agreement with existing implementations in the electrostatic limit.

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Density Control for High-Performance W7-X Plasmas

“Optimum Confinement” discharges performed at the device W7-AS were found to have experimental particle and energy fluxes in good agreement with the expectations of neoclassical transport theory over a significant fraction of the plasma minor radius (for r/a<0.7 to be precise). As these discharges had central electron and ion temperatures both well in excess of 1 keV at central electron densities as high as 1.1×10^{20} m^{-3} they are of particular relevance in the preparation of “high-performance” scenarios for W7 X, with the goal of maximizing the triple product $nT\tau$ of density, ion-temperature and energy confinement time. For centrally heated plasmas it is possible to show that steady-state solutions of the interdependent particle and energy balances require that particle and energy sources have (nearly) the identical ratio as particle and energy fluxes. This is of particular concern for electron cyclotron resonance heating (ECRH) as central particle refueling is extremely difficult for high-density plasmas, whether by neutral particle recycling or by pellet injection, and this is expected to greatly complicate density-profile control for high-performance W7 X plasmas heated solely by ECRH. Lacking a means of providing the plasma core with a source of particles, the only recourse is a combination of transport coefficients and thermodynamic forces which produce a negligible particle flux in this region. Assuming predominant neoclassical transport in the core of W7-X would then require the density profile to become hollow as a thermo-diffusive particle “pinch” does not occur in stellarators. Indeed, 1-D transport simulations predict that such plasmas will have rather hollow density profiles given the capabilities of Operational Phase 1.2 (OP1.2). It is therefore deemed important during OP1.2 to experimentally test the validity of such predictions since particle refueling with pellets can be improved by increasing the injection velocity (beyond the modest value of 300 m/s possible with the AUG blower gun) and/or changing the launch geometry.

W7-X Divertor Operation Optimization

W7-X island divertor performance under detachment conditions has been explored using the EMC3 Eirene code. High radiation fraction, low impurity density at the last closed flux surface and high neutral pressure in the divertor chamber are the criteria used for assessing the island divertor performance. Optimal conditions were explored in both configuration and plasma parameter space. Numerical results indicate that there is an optimum island size for reducing the impurity density at the separatrix and for increasing the neutral pressure in the divertor chamber. For fully-opened islands, radiation is located around the X-points. In this case, increasing the island size improves the neutral screening efficiency of the island SOL. Bigger islands with a remnant closed core of a confinement-relevant size lead to the formation of a radiation belt surrounding the islands, which cools down the islands and makes them transparent to neutral atoms.

It was also shown that a stronger diverting field b_r improves the island divertor performance in terms of impurity screening and neutral compression. A factor-of-2 increase in the radial magnetic field b_r via island control coils leads to a decrease of the carbon separatrix density by a factor of 3-5 and a rise of the neutral pressure in the divertor chamber by a factor of 3-6, depending on the radiation strength. These favorable effects are attributed to the increased efficacy of parallel transport processes, provided that the SOL plasma is operated in an ion-friction-dominating impurity transport regime at high SOL densities.

It is found that, for the specific island divertor geometry, appropriate impurity radiation eases the penetration of the recycling neutrals into the divertor chamber. For the divertor configurations of interest, the highest neutral pressure is achieved at a radiation fraction of 60-70%, showing a good compatibility between particle and energy exhaust up to this operating point. At higher radiation levels, the neutral divertor pressure falls with a slope steepening at radiation fractions of more than 80%. Meanwhile, the separatrix impurity density rises sharply and is accompanied by a rapid inward movement of the radiation layer toward the inner separatrix. Loss of density control and a subsequent inward shift of the radiation layer into the confinement region could result and restrict the radiation power removal capability of the edge islands.

High SOL plasma density is favourable for detachment operation. Simulations show that fewer impurities will be needed, and higher neutral pressure can be achieved, when detached plasmas are operated at higher plasma densities. Unfortunately, for a carbon-wall device like W7 X in the initial phase, the maximum operational density is limited by intrinsic carbon radiation. This limit can later be relaxed by installing a tungsten divertor. For the most interesting divertor configurations and SOL plasma parameters, the island SOL in W7 X is almost “opaque” for the recycling neutrals in the sense that they cannot really reach and refuel the core plasma. This offers a possibility for the core plasma density to be controlled independently, necessitating additional particle central fueling and thereby presenting a challenge for the island divertor to exhaust these externally injected particles.

Scenario Development

For the three values of the rotational transform at the plasma edge in W7-X, ι_b=5/6, 5/5 and 5/4, that are able to provide a proper boundary topology for island divertor operation, core-plasma scenarios for high-performance (high $nT\tau$) quasi-steady state operation with negligible bootstrap current have been developed. Based on these core scenarios, the divertor compatibility of the chosen magnetic configurations has been investigated using the VMEC/EXTENDER approach, which allows one to easily generate full fields for field line tracing as well as for SOL-transport calculations with EMC3/EIRENE. The high-iota configuration (ι_b=5/4) did not require adjustment, while for the standard-iota configuration (ι_b=5/5) a relatively large mirror field ($B_{01}/B_{00}=b_{01}=12\%$) and a slight reduction of the vacuum iota were necessary in order to increase the plasma volume at finite beta. A check of the new configuration confirmed the good neoclassical confinement properties and a neg-ligible bootstrap current for the desired experimental scenario. For the low-iota case with its extremely large mirror field ($b_{01}\approx 24\%$) the vacuum iota value had to be increased in order to position the boundary island correctly with respect to the divertor plates.

Ideal MHD

In continuation and extension of earlier work, W7-X divertor equilibria have been computationally established taking into account the geometry of the test-divertor unit and the high-heat-flux divertor. These equilibria were studied for their global ideal MHD properties. The work focused on the exploration of MHD unstable regions of the W7-X configuration space, thereby providing information for future experiments in W7-X aiming at an assessment of the role of ideal MHD in stellarator confinement. Consistently with the lack of shear stabilization and vacuum magnetic well in the low-iota case, unstable global MHD modes were shown to exist. If studied experimentally in W7-X, this scenario might clarify the question whether a vacuum magnetic well is needed for MHD stability of small-shear stellarators. A relaxation of this requirement could facilitate the optimisation of future stellarators by making the strong indentation unnecessary which was applied to the inboard side of the bean-shaped cross section in W7-X in order to improve instability. The standard high-mirror case is stable against low-mode-number modes. For higher plasma-beta, ballooning-type, medium-mode-number, unstable free-boundary perturbations exist according to the computational global mode analysis. This W7-X scenario could be useful to experimentally study the question whether such MHD modes might affect the plasma edge region of small-shear stellarators in a dangerous way.

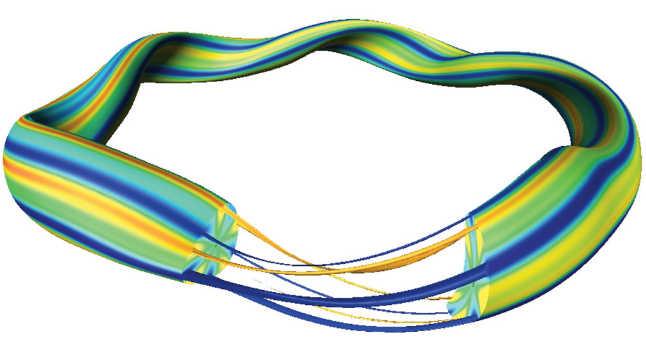

Figure 3: Mode structure of an MHD instability in a low-iota W7-X configuration.

3D Magnetic Equilibria

The Stepped-Pressure Equilibrium Code (SPEC) was brought to IPP in order to study ideal and relaxed 3D MHD equilibria. First, based on the recent conjecture that general 3D equilibria must present zonal current sheets (i.e., with nonzero surface-average) on the resonant surfaces, it could be demonstrated that the linear and nonlinear ideal response to resonant magnetic perturbations shows incomplete shielding and is greatly amplified as β is increased (figure 4). Second, SPEC was used to carry out first calculations of stellarator equilibria with islands, starting with a rigorous verification of vacuum fields and a benchmark with Biot-Savart solutions. The next steps to be undertaken are the calculation of stellarator equilibria with finite prescribed current and pressure in free-boundary, which should provide insights into (1) the effect of bootstrap current on the formation of islands, and (2) the equilibrium β limit, which is related to the emergence of magnetically stochastic regions.

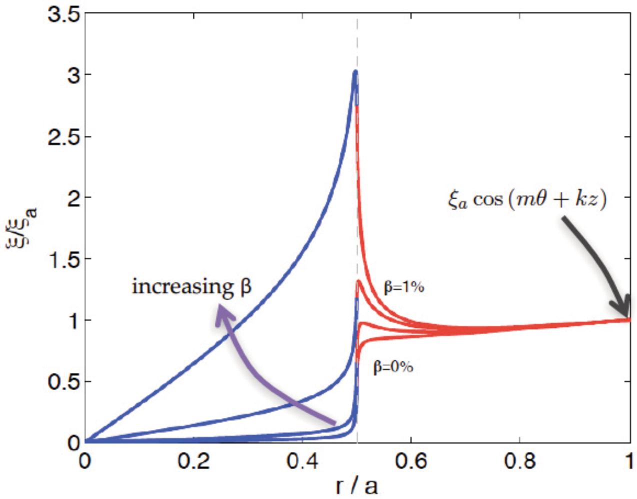

Figure 4: Ideal response due to an RMP as pressure is increased.

Neoclassical Simulations

SFINCS (Stellarator Fokker-Planck Iterative Neoclassical Conservative Solver) is a novel continuum code, which solves the radially local 4D drift-kinetic equation, retaining coupling in four of the independent phase space variables (two spatial and two in velocity). The code is unique in the sense that it simultaneously allows for 3D magnetic equilibria, an arbitrary number of species, and includes the linearized Fokker-Planck operator for self- and inter-species collisions, with no expansion made in mass ratio. The code is under continuous development, and during the last year significant additions have been made. SFINCS has been used to study neoclassical phenomena in stellarators. Specifically, the code has been used to model the neoclassical viscous torque due to non-axisymmetry in ASDEX Upgrade and the results have been compared with calculations by NEO 2. Furthermore, SFINCS was used to calculate neoclassical impurity transport coefficients for a Wendelstein 7 X magnetic equilibrium, and to demonstrate the importance of using the full linearized Fokker-Planck operator in inter-species transport calculations. SFINCS simulations have also shown that a change in plasma effective charge of order unity can affect the bootstrap current enough to cause significant movement of the divertor strike point locations. SFINCS has been benchmarked to the DKES code and to analytical predictions at high collisionality. Comparisons to DKES results for the electron root plasmas of the first Wendelstein 7-X operation phase have also been performed.

The Universal Instability

A recent publication from the University of Maryland overturned the longstanding result that the "universal" mode is stable. This achievement came as a great surprise, since it followed a long historical battle. At IPP Greifswald, new theoretical support to the numerical proof has been derived that demonstrates that a much broader class of magnetic fields in which the instability can exist, in particular the toroidal fields of fusion plasmas. Specifically, the instability called "universal" is ultimately to be found in maximum-J stellarators, where the trapped electron drive is sufficiently suppressed for the residual parallel Landau resonance to drive the universal mode. In sheared slab geometry, the universal instability can be completely stabilized with a small amount of collisional dissipation. As it turns out, this system can still be nonlinearly unstable if the magnetic shear is sufficiently weak, which could be shown by direct gyrokinetic simulation of subcritical turbulence. It could thus demonstrated that transient amplification of the gyrokinetic free energy is present, and proved that this is indeed necessary for the existence of turbulence.

Zonal Flows and ITG Turbulence

In many cases of ion temperature gradient (ITG) driven turbulence in toroidal fusion plasmas, strong zonal flows are observed and play a dominant role in the saturation of linear instabilities. The zonal flows are described as a "regulator" of the turbulence, whereby the relative amplitude of turbulence and zonal flows is set by a balance of nonlinear generation and decay of the zonal flows. A novel cascade model of this turbulent balance has been developed, and detailed test of its predictions could be verified by gyrokinetic simulations. In particular, a saturation rule for the turbulence could be derived that agrees with observed scaling of the turbulent heat flux. Even far above marginal stability, zonal flows can regulate the turbulence, despite having a diminished amplitude relative to the turbulence, confirming that the model applies for a broad range of parameters.

2D Drift Kinetic Model for ITG Turbulence

Why does plasma turbulence sometimes retain the features of the linear instabilities that drive it? What determines whether or not turbulence will exist in the absence of linear instabilities? These fundamental questions have motivated a simple limit of the electrostatic gyrokinetic system in which the linear modes are drift-kinetic and nonlinear zonal flow dynamics is determined according to the long-wavelength quasi-two-dimensional limit of gyrokinetics. For this system it could be proved that linear stability implies nonlinear stability. Furthermore it was found that the nonlinear E×B shearing preserves the identity of linear eigenmodes, preventing energy exchange between stable and unstable modes. This leads to the derivation of a dimensionally reduced model system that may prove useful in studying weakly driven turbulence (near the nonlinear critical gradient of Dimits).

Linear ITG Instability

The presence of a neoclassical radial electric field is an important feature of stellarators. Depending on the operational regime, this field either points inward (so-called ion-root regime) or outwards (electron-root regime). Both cases are of interest, but the ion root regime is most prevalent in experiments and was therefore incorporated in a series of linear ITG simulations. These simulations were designed to supplement earlier of linear ITG properties in both W7-X and LHD. The analysis of mode frequencies, power spectra, spatial structure and localization helped to clarify the dependence of the growth rate as a result of competing stabilizing mechanisms. The results generally support the common view that radial electric fields usually lead to a reduction of ITG instability growth. These fields give rise to an E×B-drift which can lead to a spatial displacement of ITG mode structures and a simultaneous modification of characteristic mode properties. Due to their different geometric properties, W7-X and LHD are seen to exhibit qualitatively different ITG modes. Two different stabilising mechanisms – the influence of a poloidal drift and profile shearing – can be recognized for both W7-X and LHD, but seem to play a more or less dominant role depending on the configuration and the strength of the electric field. For W7-X with its pronounced "helical edge", the distortion of the ITG mode due to the variation of metric properties seems to be dominant. For LHD with its smoothly varying curvature on the outboard side of the torus, the spatial displacement of ITG modes into regions with reduced instability growth can explain the observed reduction of growth rates.

An electromagnetic theory of the strongly driven ion-temperature-gradient (ITG) instability in magnetically confined toroidal plasmas has also been developed. Stabilizing and destabilizing effects could be identified, and a critical β_e (the ratio of the electron to magnetic pressure) for stabilization of the toroidal branch of the mode calculated. Its scaling is $\beta_e \sim L_T/R$, where L_T is the characteristic electron temperature gradient length and R the major radius of the torus. A fast-particle population can cause a similar stabilization due to its contribution to the equilibrium pressure gradient.

Intrinsic Turbulence Reduction in a Stellarator

By massively parallel direct numerical simulations, two effects have been discovered that lower turbulence levels in stellarators, but are absent in tokamaks. The first effect stems from the fact that turbulence in stellarators tends to localize in narrow band-like regions in the neighborhood of the most unstable field line segments. As a result, the average transport across a flux surface is much less than the peak levels. This is in contrast to tokamak turbulence, where all field lines are identical due to toroidal symmetry, and so contribute equally to the surface-average transport. One significant consequence of this effect is an observed “upshift” in the ion temperature gradient at which the onset of turbulence is observed (see figure 5). This upshift is unique to stellarators, and distinct to the Dimits shift, observable in both tokamaks and stellarators. A second stabilizing effect is found above the turbulence threshold, and is related to the suppressive interaction between large turbulent eddies, and small-scale features in the magnetic field. A theoretical explanation is proposed involving linear stabilization of the largest-scale modes. As these modes are the most responsible for transport, their suppression is associated with a reduced “stiffness” or steepness in heat flux versus temperature gradient, as compared with local numerical simulations that exclude the effect by design.

This is also demonstrated in the figure, as the transport at the single field line (yellow line, $\alpha=\pi/3$) is greater than the two surface averaged amounts, with the large ρ^* case exhibiting the greatest interaction, and therefore the greatest suppression of turbulence. The yet-to-be built QUASAR stellarator is highlighted to demonstrate these effects, but the causes are quite generic, and we therefore believe that all stellarators should enjoy the associated intrinsic turbulence stabilization.

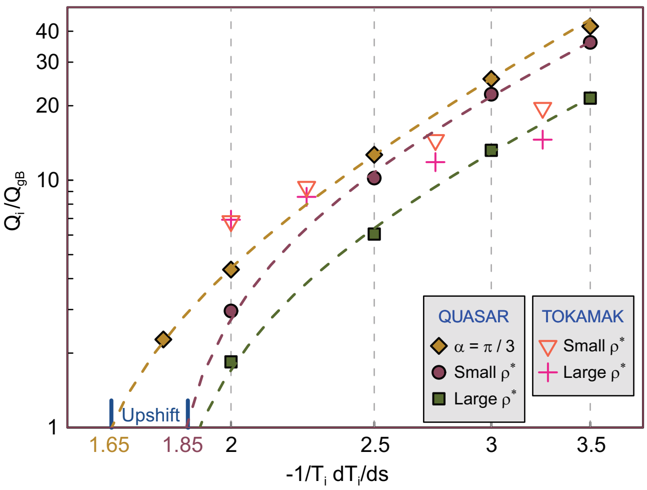

Figure 5: The transport at the single field line (yellow line, $\alpha=\pi/3$) is greater than the two surface averaged amounts, with the large ρ^* case exhibiting the greatest interaction, and therefore the greatest reduction of turbulence.

Turbulence Simulations with Kinetic Electrons

In fully nonlinear simulations with the flux-tube version of the GENE code, heat fluxes caused by ITGs and trapped-electron modes (TEMs) in DIII-D, HSX and W7-X have been compared. For ITGs, the heat fluxes were comparable in all three devices if the heat flux in both stellarators was multiplied by a factor to account for the difference in surface-to-volume ratio. In spite of having this factor, the TEM heat fluxes were significantly smaller (by about two orders of magnitude) for both stellarators compared with DIII-D. There were a couple of surprising observations that have not been explained yet. First, the growth rates of TEMs were comparable to those of ITGs in all three devices, but only in DIII-D did this lead to a comparable (and very high) heat flux. In both stellarators, the TEM-driven heat flux was much lower than the ITG heat flux. This might indicate at a difference in saturation mechanisms for ITGs and TEMs in stellarators. Second, the linear TEM growth rates of HSX were higher than those of W7-X by a factor of two, but the TEM heat fluxes were comparable. Furthermore, using the full-flux-surface version of the GENE code, the first simulations of ITGs and TEMs including kinetic electrons have been made over a complete flux surface in W7-X geometry. The growth rates found for the full surface were consistently smaller than those found in the flux tube that intersects the horizontal midplane in the bean-shaped poloidal cross section of the device, which is usually the most unstable flux tube for both ITGs and TEMs. This difference in growth rates can be attributed to an averaging effect.

Electromagnetic Gyrokinetic Simulations

It has been shown that the new algorithm used in the EUTERPE code for electromagnetic simulations, the so-called “pull-back transformation scheme” can be interpreted as an explicit time integrator reset after each full time step. A nonlinear extension, derived consistently from a field Lagrangian, was proposed and tested. The internal kink mode could be simulated in tokamak geometry using the GYGLES code. A power balance relation suitable for mixed-variable gyrokinetics has been derived and implemented in the code. A background parallel current was introduced using two approaches: through a shifted Maxwellian and with two different coordinate systems for the ions and the electrons. In the second approach, the equations of motion and the power balance for the electrons needed to be rederived since they differ from the usual formulation. The EUTERPE code has been used to study the dynamics of the Pfirsch-Schlüter currents in tokamak and stellarator geometries. The Pfirsch-Schlüter current was computed as a steady-state (equilibrium) solution of the initial value problem at a fixed pressure gradient. In tokamak geometry, the solution has been benchmarked with the analytic expression and found to be in a very good agreement. In stellarator geometry, a singular current was found to develop at resonant flux surfaces.

Fast-ion-driven Instabilities

The destabilisation of drift Alfvén modes in LHD was investigated using the EUTERPE code. It was found that a TAE-like mode with a frequency lying in the TAE gap could be destabilized by the gradients of the bulk plasma when no fast particles are present. Similar modes have also been found in W7-X. For tokamaks, the CKA-EUTERPE code package has been used to study the influence of a finite fast-particle gyroradius on mode saturation. It was found that the dominant effect is that the change in linear growth rate due to the finite gyro radius leads to a different saturation level. The dynamics of the saturation has also been investigated with a spectral diagnostic to analyse if a non-linear frequency shift (chirping) occurs. The spectral dynamics seems to be almost completely determined by the linear growth rate of the mode. For those values of the growth rate where chirping occurs, the splitting of the frequency is slightly larger if a finite gyro radius is present. Furthermore, it has been found that the saturation level scales as $B_{max}\sim\gamma^2$ for small growth rates while for large growth rates the scaling is $B_{max}\sim\gamma$. The non-linear version of the CKA-EUTERPE code has also been successfully applied to stellarators. Saturation levels of an NBI-driven TAE mode in Wendelstein 7-X and an alpha-driven HAE mode in the HELIAS reactor have been calculated. Due the more complicated particle motion, the time step necessary to converge the results is found to be smaller by more than an order of magnitude compared with a tokamak case. The saturation levels found for relevant parameters are around $B_{rad}/B_0\approx10^{-3}$.

STAE-K is a fast numerical tool that has been developed to solve the eigenvalue problem for fast-ion-driven modes using a Riccati shooting method. Using this tool, the resonant interaction of shear Alfvén waves with energetic particles was investigated in both tokamak and stellarator geometry using a non-perturbative MHD-kinetic hybrid approach, with a large-aspect-ratio-expansion for the plasma equilibrium. While the background plasma is treated within the framework of ideal-MHD theory, the drive of the fast particles, as well as Landau damping of the background plasma, is modelled using the drift-kinetic Vlasov equation without collisions. The code is sufficiently fast that it is suitable for parameter scans. When the pressure of the energetic particle is comparable with the bulk pressure, Energetic particle modes (EPMs) can be found. To reveal the physical effects underlying an EPM, the deformation of the Alfvén continuum in the presence of energetic particles was investigated, and it could be shown that the TAE frequency can leave the continuum gap. To account for damping effects such as radiative and continuum damping, and to eliminate the singularities of ideal MHD, the parallel electric field and FLR effects were accounted for perturbatively.

The FLU-EUTERPE code has been extended to work non-linearly and applied to an ITPA benchmark case for a tokamak. With a finite resistivity, a saturation of the mode amplitude close to the saturation level of the nonlinear kinetic-MHD perturbative hybrid model CKA-EUTERPE, and the similar non-perturbative hybrid code HMGC has been found. The scaling of saturated amplitude with changing linear growth is also comparable. Proof-of-principle simulations within a limited range of numerical parameters have been performed in W7-AS, LHD, and W7-X stellarator geometries. Global modes driven by a fast particle population have been found in W7-X geometry. Work is in progress to expand the parameter range to experimentally relevant values. Finally, in collaboration with the Australian National University, continuum damping in stellarators using the full geometric information and an arbitrary number of Fourier modes has been addressed by generalising an earlier developed contour integration method for a two-mode model in the large-aspect-ratio approximation.

Full-wave Calculation of RF Wave Propagation

A new 3D full-wave code with a cold-plasma dispersion relation has been developed. The code takes advantage of massively parallel computations with graphics processing units (GPUs). Modern GPUs allow for an acceleration of the calculation by two orders of magnitude compared with single CPUs. The code allows for fast and efficient calculations for ECRH heating and reflectometry, including such effects as mode conversion and mode coupling.

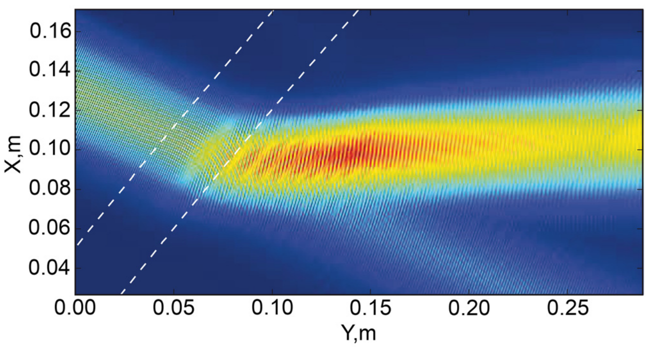

Figure 6: ECRH wave entering the plasma in W7-X and splitting into two components (X2 and O2).

Evolution of the Wave-mode Purity along the RF Beam

In order to calculate the wave-mode purity along the reference ray and estimate a possible reduction of absorption in strongly non-uniform plasmas in sheared magnetic fields, a quasi-1D model for solving the wave equations along the reference ray for coupled O- and X-modes was developed and implemented in the ray-tracing code TRAVIS. The model also includes bi-refringence, which can be significant obliquely launched beams with imperfect polarization matching with respect to the plasma periphery. The model was benchmarked successfully against the 3D full-wave code and applied for simulation of multi-pass scenarios with O2-mode launch. It was shown that the loss of mode purity at re-entrance of the reflected RF beam from the mirror leads to a reduction of the power by about 10%.

ECE Diagnostic Web Service

The ECE diagnostic module of the ray-tracing code TRAVIS has been implemented as a Web service. The client computer can remotely (over the internet) call the ECE service from within C, Java, Python or Matlab environments. The service implements the forward function for the ECE diagnostic. This means that, for given plasma profiles and magnetic configuration, the ECE service calculates the expected radiative-temperature signals at the ECE receiver and provides flux-surface mapping. The goal is to find acceptable agreement between the assumed and measured temperatures by iterating the plasma profiles and the magnetic equilibrium.

VMEC/EXTENDER-fields

Magnetic fields for full-field calculations are necessary for SOL-transport simulations or NBI-deposition calculations taking into account re-entering ions. A way to use VMEC calculations has been developed in order to generate these fields in combination with the EXTENDER webservice. A set of python-codes/scripts allows the generation of different formats, e.g. a gridded field for use with the field-line tracer/diffusion webservice. The set of codes has been used in divertor-compatibility studies for W7-X but also in stellarator reactor studies to test changes in the magnetic configuration at finite beta for reactor candidates.

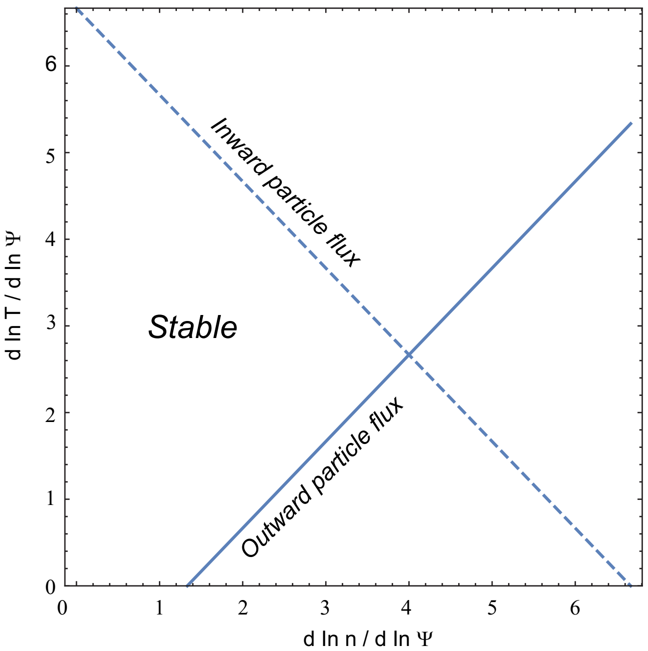

Figure 7: Stability diagram for an electron-positron plasma confined by a dipole magnetic field. The axes refer to the logarithmic density and temperature gradients, respectively. Electrostatic modes are unstable below the solid line, and electromagnetic modes above the dashed line. In the vicinity of the stability boundary, the quasilinear particle flux is inward if $d \ln T / d \ln n > 2/3$.

Electron-positron Plasmas

A consistent gyrokinetic theory for the stability of electron-positron plasmas has been developed for the parameter regime most likely to be relevant for the first laboratory experiments (underway in Garching) involving such plasmas, where the density is small enough that collisions can be ignored and the Debye length substantially exceeds the gyroradius. Although the plasma beta is very small, electromagnetic effects need to be retained, but magnetic compressibility can be neglected. It was found that gyrokinetic instabilities are completely absent if the magnetic field is homogeneous: any instability must involve magnetic curvature or shear. Furthermore, in dipole magnetic fields, the stability threshold for interchange modes with wavelengths exceeding the Debye radius coincides with that in ideal MHD. Above this threshold, the quasilinear particle flux is directed inward if the temperature gradient is sufficiently large, leading to spontaneous peaking of the density profile. Furthermore, first simulations of ITG-like instabilities in electron-positron plasmas including Debye shielding have been carried out using ORB5. It was found that this shielding is stabilising and may lead to a substantial reduction of the turbulent fluxes.

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Numerical Methods in Plasma Physics

Head: Prof. Dr. Eric Sonnendrücker

The division "Numerical Methods in Plasma Physics" is devoted to the development of efficient and robust computational methods and algorithms for applications in plasma physics and more specifically for the models and problems of interest to other divisions of IPP.

Structure of the Division

The emphasis of the division lies on the development, optimization and analysis of numerical methods and is tightly coupled with the group "Numerical methods in plasma physics" at the Mathematics Center of the TU Munich. In addition to inventing some new methods specifically for the problem at hand, the division aims at maintaining knowledge of state of the art methods in the general area of numerical mathematics and scientific computing in order to be able to adapt them where needed to plasma physics problems. The division consists of five research groups: Kinetic Modelling and Simulation, Magnetohydrodynamics, Geometric and structure preserving methods, Plasma-Material Modelling and Foundations, Zonal Flows and Structure Formation in Turbulent Plasmas. Moreover the EUROfusion High Level Support Team (HLST) is attached to the division.

Kinetic Modelling and Simulation

The group develops new algorithms and optimized implementations for the solution of the gyrokinetic equations as well as the fully kinetic Vlasov equation, considering both Particle-In-Cell (PIC) and semi-Lagrangian methods. Most of the algorithmic developments are realized within the SeLaLib library. Moreover, the analysis of various approximations in the models, verification and benchmarking of gyrokinetic and kinetic codes are an essential part of the work.

Field-aligned Interpolation for Gyrokinetics

Turbulence simulations based on the 5D gyrokinetic equations are computationally expensive because the thermal ion Larmor radius must be resolved. The computational burden can be reduced by taking advantage of the fact that the gradient of the solution is much smaller along the direction parallel to the magnetic field than in a direction perpendicular to it. The traditional solution is to align one coordinate with the local magnetic field line, but this approach severely restricts the geometry. Our approach adapts to the semi-Lagrangian context a method developed by Hariri and Ottaviani, where field-aligned differentiation (or interpolation) is performed between poloidal planes: such local field-alignment is very flexible with respect to the geometry and the spatial discretization, and it reduces considerably the number of poloidal planes in our computational mesh. In the last two years, we have extended our prototype field-aligned code for a reduced drift-kinetic model to include missing non-linear terms, so that a full 4D drift-kinetic model is now implemented in SeLaLib. As a result, in all splitting directions we solve non-constant advection equations, and the field-aligned advection algorithm was extended to accommodate a general magnetic field configuration and a 2D parallel velocity profile. This code was verified in the linear regime of an ITG instability against the results from a numerical dispersion relation analysis. Building on the experience gained from the screw-pinch model above, a more general 5D gyrokinetic framework is under development.

High-performance Semi-Lagrangian

Together with the MPCDF, we have developed a high-performance implementation of a semi-Lagrangian solver of the Vlasov-Poisson model. This solver is based on a dimensional splitting and successive 1d interpolations on stripes of the domain. Single core performance has been improved by vectorization of the interpolation step. OpenMP parallelization and cache optimization avoiding large strides in the array access have improved the single node performance. For distributed memory parallelization, two parallelization strategies have been considered: A remapping strategy that works with two different layouts keeping parts of the dimensions sequential and a classical partitioning into hyper-rectangles. Strong scaling results have been performed that demonstrate better scaling curves for the partitioning compared to the remapping strategy.

PIF for Vlasov-Poisson

The Particle-in-Cell (PIC) method couples particles to a grid in order to solve the Poisson equation, which provides the electric field needed for the transport. The particles are used as Monte Carlo samples, which introduces statistical noise that leads to artificial diffusion. We introduced a new diagnostics for the solution quality by measuring this noise in the form of variance and propagating it through the field. The variance of the conserved quantities in geometric integration provides an a posteriori estimate for the discretization error. We could also show that Fourier filtering is a noise reduction technique and when it is the most effective. Instead of using finite element PIC, conserving only energy, or finite difference PIC, conserving only momentum, we introduced Particle-In-Fourier (PIF), which conserves energy and momentum. PIF couples the particles directly to a spectral basis such that Fourier filtering is already built in. The simple structure and computational efficiency of PIF allows for better theoretical investigation of the particle method, especially with respect to stochastic aspects.

Gyrokinetic Theory and Verification

Our group is involved in the EUROfusion Enabling Research project "Verification of global gyrokinetic codes and development of new algorithms for gyrokinetic and kinetic codes". On the numerical side, we have successfully completed the linear part of the inter-code benchmark including several Particle-In-Cell, Eulerian and semi-Lagrangian codes.

On the theoretical side, we are building a hierarchy of the code models from the derivation of a parent model issued from the systematic geometrical dynamical reduction procedure. We are considering the gyrokinetic reduction procedure, which is using advanced geometrical tools, such as Lie-transform perturbation techniques and Action Principle formulation of dynamics. So far the equations implemented in the Particle-In-Cell ORB5 code have been thoroughly analyzed and related to the results of the analytic reduction and the GENE code model is currently under investigation.
Gyro-transformations (GTs) are critical for the dimensional reduction of the Vlasov equation in order to enable numerical simulations in a reduced phase space. We worked on a rigorous justification of a GT: The GT is carried out directly on the Lagrangian, systematically decoupling the fast gyro-motion from the slower gyro-center dynamics, at every order in the expansion. Moreover, Maxwell's equations are expanded in the same scaling parameter, allowing for a consistent gyro-kinetic theory in the drift ordering up to second order. Our algorithm for computing the Lie transforms composing the GT is new, and particularly easy to understand also for non-specialists.

Magnetohydrodynamics (MHD)

The MHD group works on the study, development and analysis of robust and efficient algorithms for numerical simulations of different MHD-models.

Discontinuous Galerkin Methods

The FLEXI solver implements the high order Discontinuous Galerkin Spectral Element Method on arbitrary shaped 3D hexahedral meshes. It is fully MPI parallelized and scales linearly up to thousands of cores. An interface to VMEC data has been implemented and is fully functional, providing 3D curved meshes together with the MHD equilibrium data, for both Tokamaks and Stellerator configurations. A successful MHD simulation of a 3D internal kink mode instability was achieved, for a circular cross section, using an explicit time integration (scalable, running on more than 1000 MPI ranks). A matrix-free implicit time integration has been implemented for the anisotropic diffusion equation, in order to overcome the limitations of the explicit time scheme. An anisotropic heat flux has been implemented in the full MHD equations with a sharp explicit time step estimation, considering both anisotropy from the diffusion tensor and the mesh. An aligned DG method has been successfully used to solve the heterogeneous anisotropic diffusion problem.

Jorek-Django Framework

Within the CLAPP software framework, Jorek-Django provides a set of libraries to be used for Finite Elements (and Collocation) methods. It includes a library for IO (CLAPPIO), NURBS/B-Splines for CAD-Computer Aided Design (SPL), linear algebra (PLAF), discretizations (DISCO) and an assembler library (FEMA) that allows us to combine all the different libraries and build complex models. Different models have been implemented, including a 3D reduced MHD model. Among the discretizations provided by the framework, we have cubic Hermite-Bézier elements (1D and 2D), B-Splines/NURBS and Fourier (including a spectral/collocation method). The use of Box-Splines (Splines on regular triangulations) is in progress, while Powell-Sabin elements are planned for the next year. H(div) and H(curl) approximations are also offered, thanks to the discrete exact DeRham sequence, but is up to now limited to B-Splines. All these libraries are parallelized using MPI. Different tests show a scalability of 80-94% depending on the dimension and the space discretization. Recently, we started a hybrid OpenMP/MPI parallelism. OpenACC is another option that we may consider in the future. This is a joint work with INRIA Sophia-Antipolis, INRIA Nancy Grand-Est, University of Nice and University of Strasbourg. This project has also received the support of the Bavarian Competence Network of Technical and Scientific High Performance Computing (KONWIHR).

Fast and Robust Solvers and Preconditioners

It is a known fact that classical preconditioners fail to converge in realistic viscous-resistive MHD simulations of instabilities. For this purpose, two strategies are being developed. First, we studied a method known as "physics based preconditioning" for the wave equations. It consists of approximating the solution by suitable smaller and simpler systems. Numerical simulations were conducted on Hermite-Bézier elements and B-Splines discretization. When the classical preconditioners fail to converge, our method still converges after a few iterations (1 to 10, depending on the discretization and the configuration). The second method that we are studying is a new class of preconditioners, based on the Generalized Locally Toeplitz theory (GLT). This allows us to understand the spectral pathology of the obtained matrices using a Finite Element discretization. Using this knowledge, we can derive fast and optimal solvers or preconditioners.
For the case of H^1 elliptic problems, a post-smoother for B-Splines has been implemented. It can be used with a Multigrid solver in order to remove the high frequencies due to the use of high order B-Splines. In the case of H(curl) and H(div) elliptic variational problems, we were able to derive and study the symbol of the resulting matrices. Such problems arise in Time Harmonic Maxwell and magnetostatic problems, as well as in the preconditioning of MHD equations. A detailed spectral analysis was performed for the 2D case; several numerical evidences were obtained, confirming our theoretical findings. The construction of an optimal solver is work in progress.

Geometric and Structure Preserving Methods

The group specialises in discretisation methods for ordinary and partial differential equations that preserves the structure of the original model, thus yielding stable and long time accurate simulations tools.

Geometric Discretization Methods

One of the prevalent discretization paradigms for gyrokinetics and the Vlasov-Maxwell system is the particle-in-cell (PIC) method. The PIC schemes used in present codes, do usually not keep important properties of the Vlasov-Maxwell system like its Hamiltonian structure and consequentially Casimir, momentum and energy conservation. This leads to unphysical behaviour in the numerical solutions like grid heating and a violation of charge conservation. A novel framework for finite element PIC methods has been developed, based on the discretization of the underlying Hamiltonian structure of the Vlasov-Maxwell system. This framework consists of three elements: 1) a semi-discrete Poisson bracket, which satisfies the Jacobi identity, 2) a Hamiltonian splitting scheme for time integration, 3) techniques from Finite Element Exterior Calculus and spline differential forms, which ensure conservation of the divergence of the magnetic field and Gauss' law as well as stability of the electromagnetic field solver. The resulting numerical methods are gauge-invariant, feature exact charge conservation and show excellent long-time energy and momentum conservation.

A new discretization strategy for ideal and reduced magnetohydrodynamics was devised, based on a discrete variational principle applied to a formal Lagrangian and discrete exterior calculus, which was used for the discretization of the field variables in order to preserve their geometrical character. The resulting integrators preserve important functionals like the total energy, magnetic helicity and cross helicity exactly (up to round-off). As these integrators are free of numerical resistivity, the magnetic field line topology is preserved and spurious reconnection is absent in the ideal case. Only when effects of finite electron mass are added, magnetic reconnection takes place.

Relaxation Methods for MHD Equilibria

Three-dimensional (3D) MHD equilibrium solvers are essential for stellarators and increasingly important for tokamaks due to a number of effects that break exact axisymmetry. Such codes are usually based on one of the following: an iterative solution of the MHD equilibrium equations, energy minimization, and relaxation. None of those techniques is entirely satisfactory due to either limitations on the computed equilibrium or the required computational costs. It has been suggested by Morrison that relaxation methods constructed directly from the Hamiltonian structure of MHD equations, referred to as metriplectic dynamics, could improve 3D equilibrium codes. Various techniques are available to control the relaxation mechanism, e.g., double brackets, selective Casimir dissipation, Dirac's brackets. As a first step toward the development of a 3D MHD equilibrium solver, a detailed study of such metriplectic dynamics in simple cases has been performed. The aim is to identify among the various options the most promising for a realistic equilibrium solver.

Homotopy Continuation and Artificial Relaxation for Steady-state Edge Transport

The SOLPS suite of codes is extensively used for transport modeling of the plasma edge, yet its fluid-dynamics kernel requires prohibitively small time steps for convergence when the full drift-physics is accounted for. A stripped-down model has been implemented with the aim of exposing the possible causes of the problem: The E×B drift induces a nonlinear feedback mechanism between the ion density and the self-consistent electric potential. Since a large majority of the SOLPS runs aim at a description of steady states, rather than of the whole transient, we have tested some novel techniques to converge quickly to a steady state. We have focused on homotopy continuation techniques and on artificial relaxation mechanisms.

Artificial relaxation methods are essentially based upon the same idea explained above for the case of MHD equilibria, with the difference that edge transport models are non Hamiltonian and exhibit already a few dissipation mechanisms. We proposed to modify the dynamics so that some key functionals, e.g., the L^2-norm of the residual, are dissipated efficiently. The implementation of such modified dynamics in a finite element framework requires some care: In order to avoid the complication of high-regularity finite elements an appropriate splitting with compatible finite element spaces have been devised, as done in least-square finite elements. We obtained a clear speed up of the convergence to steady-state.

Compatible Finite Elements for the Quasi-neutrality Constraint in Edge Transport

Both modeling and numerical computations of the edge transport introduce various approximations aiming at removing fast modes from the plasma dynamics, either because they are considered negligible for such problems or because their effects can be represented by simpler models. One of these approximations is quasi-neutrality, which amounts to assuming that the divergence of the current vanishes. While being physically justified and computationally convenient, such an assumption significantly changes the structure of the equations, affecting in particular the way in which the electrostatic potential is determined. This is a subtle mathematical aspect that has practical implications for the development codes, such as the SOLPS suite.

An analysis of the model resulting from the quasi neutrality condition has been performed, showing its close relation with well-known models from standard fluid dynamics and pointing out the role of the electrostatic potential as a Lagrange multiplier. Following this, a consistent finite element discretization has been proposed and studied, both theoretically and numerically.

Plasma-Material Modelling and Foundations
The group has at present three main research foci: 1) The development of methods for uncertainty quantification (UQ) of computer experiments. 2) An improved modelling of plasma-wall interaction. 3) An analysis of inverse problems using Bayesian methods.

Uncertainty Quantification Using Gaussian Processes
The analysis of computer experiments poses statistical challenges that differ from the standard evaluation of experimental data. Established methods like replication and randomization are irrelevant because a (deterministic) computer model yields the same answers if run multiple times. In addition, due to the complexity of the physics model, computer simulations in the area of plasma-wall interactions are very time-consuming, such that only a limited number of simulation results are available. Therefore, already in relatively low to medium dimensional settings the quantification of the uncertainty of computer experiments is challenging. A relatively new and promising approach is the use of an emulator, a meta-model as a computationally cheap surrogate for the computer model. Gaussian processes provide a non-parametric approach for such an emulator and are suitable also for higher dimensional settings. We applied the Gaussian process method within a Bayesian framework to the uncertainty quantification of SDTRIM simulations of hydrogen isotope impact on tungsten. Employing the Kullback-Leibler-divergence as a measure of information, we established an optimized and automated sequential parameter selection algorithm to find the next best point at which it is most expedient to ask for an output either obtained by experiment or by starting the next computer run. In order to utilize the parallel running capabilities of present computer technology we marginalized over the expected outcomes at optimized test points to set up a pool of starting values for batch execution and became independent from the originally sequential procedure.

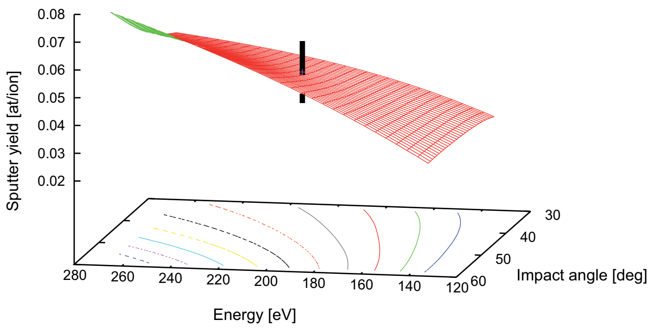

Figure 8: Sputter yield reproduced by a surrogate model from the uncertainty quantification of SDTRIM calculations for deuterium impact on tungsten with parameters of the order of E=200 eV, ESB=4.28 eV and an incident angle of 45 degrees. The corresponding contour lines are indicated on the base plane. The line in the center resembles the expected yield for the nominal parameters of 0.0497 with a standard deviation of 0.0112 (dark line).

Improved Modelling of Plasma-wall Interaction
Usage of RAFM steels like EUROFER is presently investigated as a possible option for plasma-facing components in the far-SOL layer of a future fusion reactor, the key idea being a sputtering-induced self-protection (enrichment of tungsten) of the exposed surfaces. The experimental results are promising but a code capable of modelling the coupled phenomena of sputtering, mixing and diffusion was not available. In collaboration with the stellarator theory the Monte Carlo code SDTRIM was extended to include also solid-state diffusion. The augmented code could quantitatively explain the observed data as result of a complex interplay of sample temperature, particle flux, recoil implantation and diffusivity.

Analysis of Inverse Problems Using Bayesian Methods
High heat loads onto plasma facing components, especially on the divertor or transient ones are a significant concern for the design of future fusion reactors. Nevertheless, the heat load pattern also provides significant information about the plasma transport properties in the (far-)SOL and in the divertor. The presently favoured approach to measure these heat loads are based on the evaluation of IR- and visual thermography measurements. However, the relation between the measured temperature and the actual heat load contains an exponentially smoothing kernel, which hampers the evaluation of the heat loads. Here the approach of Bayesian adaptive kernels, which has so far successfully applied to 1-d deconvolution problems has now been extended to the thermography problem and is presently applied to data from ASDEX Upgrade and JET.

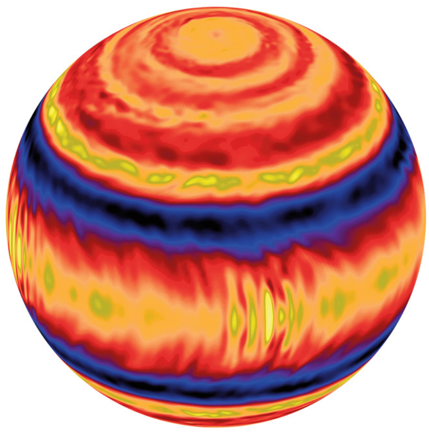

Figure 9: Zonal Flow/turbulence simulations: Colour coded zonal flow velocity on the planetary surface.

Zonal Flows and Structure Formation in Turbulent Plasmas
Zonal Flows in Plasmas and Planetary Atmospheres
The main subject of the group is the behavior of turbulence induced global flows in magnetized plasmas, and, comparatively, in planetary atmospheres. These flows – and other

comparable global structures – are vital for the prediction of the amplitude and time evolution of the turbulence. The investigations are carried out using two-fluid (NLET) and gyrokinetic (GYRO, GS2) first principles computer simulations of plasma turbulence, as well as anelastic planetary turbulence simulations utilizing the code NAN (figure 9). In the following, the focus is on the gyrokinetic and fluid results.

Zonal Flows and GAMs in Comparative Gyrokinetic and Two-Fluid Turbulence Simulations

Gyrokinetic simulations have come to be the standard of current turbulence modelling in the low gradient, nearly collisionless region in the core of tokamak discharges, since they include nearly all the kinetic effects to be expected for a small ratio of the turbulence to the Larmor frequency. On the other hand, fluid turbulence computations allow much higher spatial resolution, become more reliable at large collision-frequency, can be adapted to non-Boussinesq scenarios and tend to more directly yield physical insight rather than fully kinetic simulations. The large ratios of domain size to vortex diameter accessible by fluid simulations are also relevant to the study of deterministic zonal flow (ZF) interactions. At present the full nonlinearities that become relevant at the notoriously difficult edge region of tokamaks, due to the high fluctuation amplitudes there, cannot be implemented in a gyrokinetic framework. In addition the high collision numbers at the edge make gyrokinetic simulations numerically and physically challenging.

To extend the predictive capabilities into the fringe regions of validity of both approaches, and to isolate special kinetic effects from the more robust fluid physics, results on GAMs and ZFs obtained with gyrokinetic and fluid codes have been compared. The specific study of the global flows is of particular interest, with a view towards an eventual understanding of the L/H transition and the associated edge flows.

In non-marginal regions, far from instability thresholds, the results of the fluid code are in rather good agreement with the kinetic results, even if the collisions are scarce. This can be understood, because at sufficiently large growth rates, the resonances responsible for fine phase space structures become sufficiently wide to allow a representation by fluid moments of the distribution functions.

As an example, figure 9 shows a gyrokinetic-fluid comparison of GAM oscillations for collisionless ions and adiabatic electrons in the core/edge transitional regime. The amplitude, pattern, time- and length-scales of the flows agree rather well, indicating that kinetic effects due to higher moments of the distribution function (which are absent in the fluid code) are unimportant in this non-marginal scenario.

The comparisons show that ZFs are generated similarly by a Reynolds stress based self-amplification and interact with the turbulence modes through wave kinetic effects, while GAMs mostly result from a modulation of the background diamagnetic velocity. Several prior results from the fluid code have been confirmed with the gyrokinetic codes, such as near-deterministic flow evolution for large enough domain sizes, turbulence modulation by GAMs, and stationary behavior of slab-drift-wave generated ZFs.

An important caveat raised by the comparisons is that particular care has to be taken with the physics and numerics of the collision operator used in the gyrokinetic codes, so that the proper fluid limit is eventually reached for high collisionalities. On the other hand, the gyrokinetic results can guide the proper renormalization of the fluid dissipative terms to account for the kinetic damping mechanisms (Landau-damping and phase-mixing) to prevent a partial break-down of the fluid description at the lower collisionalities.

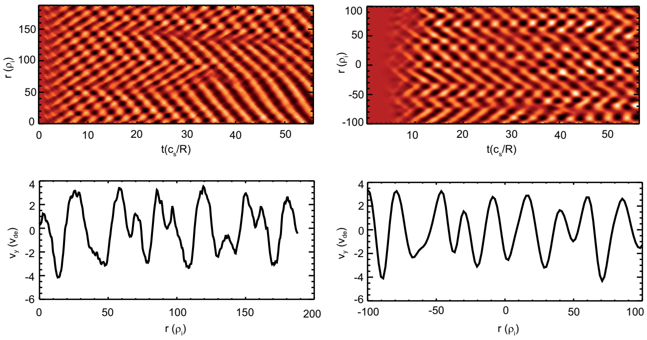

Figure 10: Comparison of GAM oscillations for collisionless ions and adiabatic electrons in the core/edge transitional regime using a gyrokinetic (GYRO, left panels) and two-fluid code (NLET, right panels). The parameters were R/Ln=10, R/LT=30, a/R=0.2, q=3.2. Upper panels: colour coded flux surface averaged flow velocity as function of minor radius and time; lower panels: snapshots of the flow profiles at t=21 and t=28, respectively.

High Level Support Team

Tasks of the High Level Support Team

The High Level Support Team (HLST) provides support to scientists from all Research Units of the EUROfusion consortium for the development and optimization of codes to be used on supercomputers. One such machine is the IFERC-CSC supercomputer called "HELIOS" located in Rokkasho, Japan. It delivers compute power of about 1.5 petaflop/s. Another one is the "MARCONI" supercomputer hosted by CINECA in Bologna, Italy with a total compute power of about 2 petaflop/s.

The HLST consists of a core team based at IPP Garching and of staff members provided by the other Research Units. The former has six members and the latter contributes with an additional three scientists.

In 2015 and 2016 the HLST core team has been involved in nine different projects submitted by scientists from all over Europe. By way of example, we present here an overview of the work being done for three of these projects.

BEUIFERC Project

We analysed the performance of the Intel MIC partition of the HELIOS supercomputer by means of different micro-benchmark tests. The network performance was tested first by means of the Intel MPI Benchmark suite. It was found that the two available Infiniband ports per node were not working properly since they were providing a lower memory bandwidth than expected. A software solution could be found resulting in a three times higher memory bandwidth. The different OpenMP pragma overheads were measured on both the Sandy Bridge node and the MIC card using the EPCC micro-benchmark suite. The overhead time is more than a factor of ten higher on the MIC than on the Sandy Bridge processor due to the large number of threads used in the former.

New features of the MPI 3.0 standard (the collective communication, the remote memory access and the shared memory segments) were also tested on HELIOS. For collective communication an efficient communication-calculation overlap (85-99%) could be achieved only with large message sizes (>10 KB), while for small messages this overlap stayed between 20 and 55%. The passive target communication in the remote memory access of the MPI 3.0 standard works properly only when the asynchronous progress support is switched on, allowing to achieve more than 95% overlap between calculation and synchronization. Finally, the implementation of the shared memory segments by means of the MPI standard 3.0 was verified to provide correct result, showing that all MPI tasks can have direct access to a shared memory region being created initially by a master task.

JORSTAR Project

In the JORSTAR project the STARWALL code has been analysed for potential improvements and optimization by means of MPI parallelization. For large production runs the entire code had to be parallelized due to local memory restriction for saving the input/output matrices and due to the computational time for several subroutines. The LAPACK subroutine for the eigenvector solver was replaced by its parallel counterpart from the ScaLAPACK library. Interestingly the ScaLAPACK subroutine has also shown better performance in sequential mode due to the advantage of using IEEE arithmetics.

Several subroutines were re-written in order to avoid the building up of too large matrices. In combination with the MPI parallelization concept this finally allows to perform simulations of ITER sized problems, i.e. to resolve the realistic wall structure by finite elements. Good parallel scalability was achieved in all modified subroutines with a speed-up factor of more than 210 when using 512 cores. The simulation time in such a case is less then 12 hours using 128 computing nodes on HELIOS. Finally, the complete code was parallelized including all LAPACK and user written subroutines. The new parallel version of the code provides identical results to the original one.

SOLPSOPT Project

The SOLPS code package is widely used to simulate Scrape-Off Layer (SOL) plasmas. Two main components of this package are the B2 and the EIRENE codes. B2 is a plasma fluid code to simulate edge plasmas and EIRENE is a kinetic Monte-Carlo code for describing neutral particles. EIRENE is parallelized with MPI, and the B2 code is parallelized using OpenMP. The SOLPSOPT project is a continuation of the PARSOLPS project, aiming to further improve the parallelization and decrease the execution time of B2 and the coupled B2-EIRENE system. More than 25 subroutines have been parallelized, and 90% of parallelism has been reached in the whole B2 code. With these changes a factor of six speed-up could be achieved for the ITER test case when executed on a single compute node. Several of the subroutines were optimized to reach a speedup, which is close to the bandwidth limit. As a positive side effect, these optimizations led to a speedup of the sequential version of the code, which is now 20% faster than originally. The individual subroutines have been tested separately with unit tests to ensure the correctness of the modified B2 code. The whole code was tested for both shorter and longer simulations, and the results agree with the original code.

Unfortunately, the MPI version of EIRENE in the SOLPS 5.0 code package was not functioning, so it was decided to switch to the latest version of EIRENE from SOLPS-ITER. The parallel performance of the stand-alone version of EIRENE was investigated, and we have found that the currently implemented parallelization strategies do not scale due to load imbalance. A simple and balanced parallelization strategy was implemented instead. Tests with different particle numbers show that it provides reasonable speedup. However, the tests revealed that the parallel version of EIRENE gives incorrect results, even if we use the original code version. The corresponding MPI bug was localized in the coupling subroutines between B2 and EIRENE. The problem was corrected, and the code calculates now correct results for the AUG and ITER test cases in parallel mode. The bug fix and improved parallelization schemes will be submitted to the ITER repository of SOLPS.

Scientific Staff

J. Ameres, C. Bressan, C. Caldini-Queiros, B. Dingfelder, A. Dodhy-Würsching, T. Fehér, M. Gaja, Y. Güçlü, L. Guidi, K. Hallatschek, R. Hatzky, F. Hindenlang, O. Hoenen, K. S. Kang, K. Kormann, M. Kraus, J. Lakhlili, O. Luk, O. Maj, M. Martone, M. Mazza, S. Mochalskyy, D. Nille, H. Oberlin, S. Possanner, R. Preuss, A. Ratnani, M. Restelli, T. Ribeiro, U. von Toussaint, N. Tronko, E. Zoni.

Long Time Guest

P. Morrison

Max Planck Princeton Cooperation

Max Planck Princeton Research Center for Plasma Physics

Heads: Prof. Dr. Sibylle Günter, Prof. Dr. Per Helander

MPPC has continued to thrive in its fourth year of operation, and has now reached a stage where collaborations, mostly between IPP and PPPL, are spreading to topics and staff that are not explicitly funded by the Center. Workshops held in Princeton and Berlin have attracted visitors from the astrophysical plasma community, and Japanese colleagues have attracted national funding to participate in the scientific exchange.

Plasma Turbulence

In the area of plasma turbulence, the MPPC facilitates collaborations between IPP, PPPL, Princeton University, UCLA, and several other partner institutions. The research focuses on fundamental aspects of plasma turbulence as well as on applications to fusion and astrophysical plasmas. We exemplify progress by summarizing two recent research highlights.

The gyrokinetic turbulence code GENE, originally developed for tokamak applications, has recently been extended to cover full-flux-surface variations of non-axisymmetric magnetic geometries and employed to study turbulence stabilization in stellarators [1,2], as described in the section on Stellarator Theory. It has also been applied to the study of turbulent dissipation in natural plasmas like the solar wind, a problem which is widely regarded as a key unsolved problem in space plasma physics. Nonlinear energy transfer and dissipation in Alfvén wave turbulence were analyzed in the first gyrokinetic simulation spanning all scales from the tail of the MHD range to the electron gyroradius scale [3]. For typical solar wind parameters at 1 AU, about 30% of the nonlinear energy transfer close to the electron gyroradius scale is mediated by modes in the tail of the MHD cascade. Collisional dissipation occurs across the entire kinetic range (below the ion gyroradius). Both mechanisms thus act on multiple coupled scales, which have to be retained for a comprehensive picture of the dissipation range in Alfvénic turbulence. Questions regarding the relevance of phenomena not contained in gyrokinetics (like cyclotron resonances) have also led to a systematic comparison between gyrokinetics, a hybrid kinetic-ion/fluid-electron approach, and a fully kinetic approach [4]. It was found that the gyrokinetic model, while lacking high-frequency solutions and cyclotron effects, faithfully reproduces the fully kinetic Alfvén wave physics close to, and sometimes significantly beyond, the boundaries of its formal range of validity.

[1] *P. Xanthopoulos, H. E. Mynick, P. Helander, Y. Turkin, G. G. Plunk, F. Jenko, T. Görler, D. Told, T. Bird, and J. H. E. Proll, Physical Review Letters 113, 155001 (2014).*

[2] *P. Xanthopoulos, G. G. Plunk, A. Zocco, and P. Helander, Physical Review X 6, 021033 (2016).*

[3] *D. Told, F. Jenko, J. M. TenBarge, G. G. Howes, and G. W. Hammett, Physical Review Letters 115, 025003 (2015).*

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Energetic Particles

Several AUG scenarios have been identified to serve as a basis for investigating the interaction between energetic particles and turbulence. Typically, both electrostatic (EGAM) and electromagnetic (TAEs, RSAEs) modes are present. In order to prepare the analysis with linear and nonlinear codes, the effect of the turbulence on the bootstrap current has been investigated. Using a turbulent extension of Ohm's law derived from a self-consistent transport theory, effects of magnetic turbulence in shaping the current density have been analysed. Besides the well-known hyper-resistive contribution, the generalized Ohm's law contains an anomalous resistivity term and a term proportional to the current density derivative. This so-called cross-resistivity generates current from the transfer of linear momentum from the turbulence to the electrons, thus amplifying the parallel current. It is a turbulent bootstrap contribution that could explain observed discrepancies between the theoretically predicted bootstrap current and the measured one. The anomalous resistivity leads to a reduction of the current density in the central part of the plasma, and to an increase in the outer region. The results confirm the possibility of negative turbulent anomalous resistivity, a fact that according to the transport model is due to the combined effect of the thermodynamic and magnetic equilibrium profiles. Hyper-resistivity flattens the plasma current in the centre of the plasma, while anomalous and cross-resistivity are more important in the edge.

Most of the turbulent bootstrap current is compensated by the anomalous resistivity, so that the increase in total current is only around 2%. Nevertheless, the current redistribution can raise the central value of q to above unity, thus suppressing the sawteeth, which could explain experimental data from AUG. The competing effects of anomalous and cross-resistivity on the total current cancel out in L-mode, but are expected to be important when gradients are stronger. An obstacle to a fully bootstrapped tokamak is that no current arises near the magnetic axis, but turbulence could help by providing inward diffusion of the bootstrap current.

Magnetic Reconnection

Tokamak Sawteeth

One of the long-standing questions in sawtooth physics is "How much flux is reconnected during the crash?". The original Kadomtsev model assumes full reconnection and leads to a central safety factor q_0=1 in the plasma core after the crash. The existence of (1,1) post-cursors in experiment contradict this simple picture. During the last AUG campaign, measurements of the q_0 changes during the sawtooth crash were made with a new Imaging Motional Stark Effect (IMSE) diagnostic, which was used together with the conventional MSE system to measure changes of q_0 during the crash in L-mode and H-mode plasmas. First results show stronger changes in q_0 in L-mode plasmas than in H-mode. More detailed interpretation of the measurements will be done with modeling using special equilibrium reconstruction code.

Experiments on VINETA II

Previous campaigns which characterised the reconnecting current sheet in VINETA II show a three-dimensional structure in which the reconnection rate varies along the X-line. The inhomogeneity of the reconnection current sheet will be amplified by superimposing an axially localized reconnection drive system forcing locally higher reconnection rates, which are expected in a guide-field situation to equilibrate along the current sheet (so-called reconnection propagation). The new 3D probe diagnostic system will be indispensable to study the resulting full three-dimensional dynamics.
Experiments to date have focused on a fast reconnection drive regime in which the magnetic flux is driven on time scales faster than the ion cyclotron time ($f_{drive} > f_{ci}$) and the current sheet is dominated by the electron dynamics. In this configuration, the current sheet is not in force balance with the magnetic pressure of the in-plane field with a strong contribution of in-plane electrostatic fields [5].
The reconnection drive has been modified to operate in the regime $f_{drive} < f_{ci}$ providing the same inductive electric field as in the low-frequency drive case. The plasma gun has been upgraded to an array of seven guns in a modular arrangement to provide a larger area and higher amplitude electron current source.
A fast swept Langmuir probe ($f > f_{drive}$) has been developed which has been used for characterization of the in-plane force balance of the current sheet. Detailed time-resolved measurements of the plasma density, electron temperature and plasma potential enable an assessment of the full force balance including $j \times B$ terms known from magnetic measurements as well as electrostatic fields. These measurements show the presence of strong radial electric fields which determine the current sheet stability and will be continued to accompany the changes in operational regimes to ion time scales.

Modeling of the Reconnecting Current Sheet on Electron and Ion Scales

Within the Max-Planck Princeton Center collaboration, the Max-Planck Institute for solar system research (MPS) has been supporting the experimental campaigns by supplying two complementary computer simulation codes. A 3D EMHD model has been applied using the physical parameters of the experiment which reflects the previous experimental situation where $f_{drive} > f_{ci}$ and the current sheet width is comparable to electron spatial scales. Here, the overall spatial structure of the current sheet and, more importantly, basic features of previously observed broadband magnetic fluctuations that correlate with the local current density have been identified (figure). The same model has been previously used to simulate propagation of reconnection, and will prove a useful tool to identify useful parameter regimes and dependencies once those experimental campaigns are underway.
In addition, development of a 3D hybrid model incorporating ion motion through a PIC code enables the inclusion of ion dynamics, and is planned to accompany experiments on the drive timescale $f_{drive} < f_{ci}$.

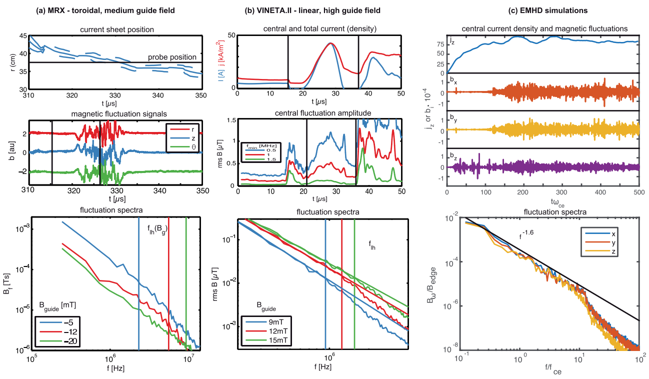

Figure: Comparison of magnetic fluctuation characteristics in the experiments [(a) and (b)] and EMHD simulations (c). Top: current sheet position and local current density. Middle: magnetic fluctuation signals and amplitudes. Bottom: resulting broadband fluctuation spectra.

Collaboration with MRX (PPPL) on Guide Field Reconnection and Magnetic Fluctuation Dynamics

A second joint experimental campaign was carried out as part of the MPPC collaboration on the MRX experiment, investigating the medium-guide-field regime in terms of overall reconnection dynamics and magnetic fluctuations. Detailed spatial plasma pressure profiles have revealed the postulated quadrupolar structure that contributes to fast reconnection by parallel pressure gradients [6]. The magnetic fluctuations show similar features to those observed in VINETA.II (figure), displaying a spectral break near the lower hybrid frequency.

Scientific Staff

P. Astfalk, A. Banon Navarro, V. Bratanov, S. Cerri, I. Chavdarovski, N. Fahrenkamp, T. Görler, O. Grulke, D. Groselj, P. Helander, V. Igochine, F. Jenko, P. Lauber, D. Milojevic, G. Plunk, J. Proll, I. Shesterikov, D. Told, A. von Stechow. M. Weidl, P. Xanthopoulos.

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University Contributions to IPP Programme

Cooperation with Universities

Author: Gregor Neu

Teaching and Training

IPP is highly interested in fostering national and international students' interest in high-energy plasma physics and other fusion-relevant fields like plasma-material interaction. This interest is reflected in the long-term endeavour of teaching plasma physics at various universities in Germany and abroad.

Many important goals in plasma physics, technology and materials science have to be attained on the way to a fusion power plant. Since this process will last another generation, IPP attaches great importance to training young scientists. Close interaction with universities in teaching and research is therefore an important part of IPP's mission. Moreover, joint projects with several universities form an integral part of IPP's research programme.

University	Members of IPP staff
University of Greifswald	Dr. Hans-Stephan Bosch Dr. Andreas Dinklage Prof. Olaf Grulke Prof. Per Helander Prof. Thomas Klinger Dr. Heinrich Laqua Prof. Thomas Sunn Pedersen
Techn. University of Berlin	Prof. Robert Wolf
Techn. University of Munich	Dr. Gregor Birkenmeier Prof. Sibylle Günter Dr. Klaus Hallatschek Dr. Michael Kraus Dr. Philipp Lauber Dr. Peter Manz Prof. Rudolf Neu Prof. Eric Sonnendrücker Prof. Ulrich Stroth
University of Munich	Dr. Thomas Pütterich Dr. Jörg Stober Prof. Hartmut Zohm
University of Augsburg	Prof. Ursel Fantz Dr. Marco Wischmeier
University of Ulm	Dr. Thomas Eich Dr. Emanuele Poli Dr. Jeong-Ha You
University of Bayreuth	Dr. Wolfgang Suttrop
University of Stuttgart	Dr. Alf Köhn
Techn. University of Graz (AT)	Dr. Udo v. Toussaint
Techn. University of Vienna (AT)	Dr. Matthias Willensdorfer Dr. Elisabeth Wolfrum
University of Gent (BE)	Prof. Dr. Jean-Marie Noterdaeme
Waseda University (JP)	Dr. Michael Kraus

Table: IPP staff who taught courses at universities in 2015 and 2016.

During the reporting period (WS 2014/15 – SS 2016), 32 members of IPP taught at universities or universities of applied sciences: Many of the IPP staff are Honorary Professors, Adjunct Professors or Guest Lecturers at various universities and give lectures on theoretical and experimental plasma physics, fusion research, data analysis and materials science. The table gives an overview. The teaching programme has been highly successful over the years and many students who first came into contact with plasma physics through lectures given by IPP staff have later done thesis work and even taken up a career in fusion research.

Lecturing at universities is supplemented by IPP's very popular "Summer University in Plasma Physics": one week of lectures given by IPP staff and lecturers from partner institutes providing detailed tuition in nuclear fusion – in 2015 in Garching and 2016 in Greifswald for the 30th time. Most of the about 70 participants per event are from Europe but there is also a number of attendees from abroad.

IPP takes an active part in the "Joint European Research Doctorate in Fusion Science and Engineering and European Doctoral Network" in the frame of which a "European Doctorate" title is awarded to PhD students in parallel to a conventional one. Obtaining the title requires spending a significant part of the work on their subject at another European university or research centre. The programme was initiated in 2008 and is presently supported by institutions in Germany, Italy (University of Padua and University of Naples "Federico II"), Portugal (Instituto Superior Técnico) and Finland (Tampere University of Technology). With the organisation of the yearly "Advanced Courses in Fusion" IPP provides a major contribution to this programme.

IPP is one of the nine full partners of the "Joint Doctoral College in Fusion Science and Engineering (FUSION-DC), which has been approved under the auspices of Erasmus Mundus, the European programme to promote training schemes. The doctoral college founded in October 2011 is being supported with about five million Euros and provides 40 doctoral scholarships for work in the field of fusion research. IPP is also an associate member of the Erasmus Mundus "European Master of Science in Nuclear Fusion and Engineering Physics" programme.

When the decision was made to build ITER, it became clear that training of young scientists and engineers had to be intensified. A European Fusion Education Network (FuseNet) was therefore formed in the 7th EU Framework Programme (FP7, 2007-2013). FuseNet currently consists of 14 research organisations – one of them IPP – 37 universities from 22 European countries and 8 industrial and other organisational members. Throughout the 8th Framework Programme, Horizon 2020 (2014-2020), FuseNet is funded through the EUROfusion consortium.

The highly international character of fusion related education at IPP is also reflected in the countries of origin of IPP's graduate students: about half of the postgraduates and approximately two-thirds of the postdocs are from abroad. In the years 2015 and 2016 a total of 88 postgraduates were supervised, 19 of them successfully completing their theses.

Joint Appointments, Grown and Growing Cooperation

IPP closely cooperates with universities in joint appointment programmes: three W3 appointments at the University of Greifswald, a W3 and a W2 appointment at the Technical University of Berlin, and two W3 and a W2 appointment at the Technical University of Munich, mark the successful implementation of these programmes. A similar arrangement exists with the University of Augsburg, where an acting IPP department head holds a W2 position and leads a research group in experimental plasma physics.

IPP entertains strategic cooperations with the Department of Theoretical Physics IV at the University of Bayreuth (pp 113), the Technical University of Berlin with two research groups in Plasma Physics and Plasma Astrophysics (pp 115), and the "Institute of Interfacial Process Engineering and Plasma Technology IGVP" of the University of Stuttgart. Other long-term cooperations described in this chapter are that with the University of Augsburg (pp 111) and the IShTAR project (pp 117), which involves the Universities of Ghent (BE) and the Universities of Lorraine and of Aix-Marseille (FR).

Networking

In addition, IPP uses specific instruments developed by the Max Planck Society, the Helmholtz Association, Deutsche Forschungsgemeinschaft (DFG), Leibniz-Gemeinschaft or the German government for more intensive networking with universities on a constitutional basis – partly in conjunction with non-university research partners and industrial partners.

Organisation of or participation in graduate schools:
- The International Helmholtz Graduate School for Plasma Physics (HEPP), started in October 2011, which is a graduate school for doctoral candidates at the Max-Planck-Institute for Plasma Physics (IPP) and their partner universities the Technical University of Munich (TUM) and the Ernst-Moritz-Arndt University of Greifswald (EMAU). Associated partners are the Leibniz Institute for Plasma Science and Technology (IPN) in Greifswald and the Leibniz Computational Center (LRZ) in Garching. HEPP aims to provide a coherent framework at IPP and the participating universities for qualifying a new generation of internationally competitive doctoral candidates in the field of plasma physics, fusion research, computational physics, and surface science.

Research partnerships:
- Until June 2015, IPP was involved in the DFG Research Training Group on "Intermolecular and Interatomic Coulombic Decay", together with the Goethe-Universität Frankfurt, Universität Innsbruck, Universität Heidelberg, Universität Hamburg, and Helmholtz Zentrum Berlin. The research unit focusses on the investigation of a mechanism for the transformation of electronic energy created by excitation or ionization with radiation in the UV and far beyond, or with energetic particles.

Helmholtz Young Investigators groups:
- The goal of the group on "Macroscopic Effects of Microturbulence Investigated in Fusion Plasmas" led by Dr. Rachael McDermott is to provide a better fundamental understanding of the interactions between the different turbulent transport channels in fusion plasmas. The group will run until December 2017. Its partner is the University of Augsburg.
- The group on "Particle transport in high temperature plasmas" led by Dr. Benedikt Geiger aims at understanding the mechanisms guiding the transport of impurity ions and of suprathermal ions in fusion plasmas. The group runs since May 2016 and will be funded until 2021. The group collaborates with the University of Greifswald.
- Recently (October 2016) Dr. Daniel Told was awarded the Helmholtz Young Investigator Group on "Hybrid gyrokinetic computations for weakly magnetized plasmas in nature and the laboratory". In close cooperation with the University of Bochum, the group will investigate turbulent heating and transport in weakly magnetized plasmas such as those present in the edge region of fusion experiments.

Virtual Institutes
- Helmholtz Virtual Institute "Plasma Dynamical Processes and Turbulence Studies using Advanced Microwave Diagnostics" where IPP cooperates in basic research of plasma-dynamics and the development of novel microwawe diagnostics with the University of Stuttgart, the Technical University of Munich, the École Polytechnique, Palaiseau (F), the École Polytechnique Féderale de Lausanne (CH), the University of York (GB), and the Plasma Science and Fusion Center of the Massachusetts Institute of Technology (US).

Universität Augsburg
AG Experimentelle Plasmaphysik (EPP)

Head: Prof. Dr.-Ing. Ursel Fantz

The research at the University of Augsburg focusses on diagnostics of low temperature plasmas, on investigations of the plasma chemistry in molecular plasmas and on plasma surface interaction. For that purpose, several different low pressure plasma experiments are available. Fundamental studies for the development of negative hydrogen ion sources for ITER and DEMO are carried out in close collaboration with the ITER Technology & Diagnostics Division of IPP.

Developments for Negative Hydrogen Ion Sources

One of the challenges in caesiated negative hydrogen ion sources for achieving stable and reproducible source performances is the caesium dynamics in these sources. For systematic studies, a planar ICP plasma, which has similar vacuum and plasma conditions as the IPP sources, has been equipped with multiple diagnostics. In particular, the set-up allows for investigations on the work function of caesiated surfaces as well as of alternatives to caesium. Recently a cavity ring-down spectroscopy system was added to measure the negative ion density above the surface with the aim to reveal for the first time the correlation between the work function and the negative ion density in the hydrogen plasma. The system has been successfully tested for purely volume generated negative ions; very preliminary results of the correlation with a caesiated surface have been obtained. Systematic investigations will be complemented by a measurement of the caesium density via tunable diode laser absorption spectroscopy which is in preparation.

Caesium-free alternative materials for negative hydrogen ion formation are investigated in view of a DEMO NBI system at a flexible ECR discharge operating at ion source relevant conditions. The effect of different converter materials on the negative hydrogen ion density is measured by means of laser photodetachment. Investigated materials are categorized according to their effect on the H^- yield: (i) tantalum and tungsten which are both expected to enhance the population of vibrationally excited molecules and thus the production of negative ions via processes within the plasma volume; (ii) LaB_6 and lanthanum-doped molybdenum (99.3 % Mo, 0.7 % La; MoLa) as low work function materials; (iii) diamond-like materials with and without dopants (boron) showing negative ion enhancement due to their negative electron affinity. The materials have been examined under the same conditions to ensure comparability. Stainless steel was used as reference for volume formation only and an in-situ caesiated surface as the target to be achieved. Figure 1 shows the obtained results. In contrast to investigations performed with evaporated materials in the literature, the used bulk samples of Ta and W do not show an enhanced negative ion yield. The carbon based materials also do not enhance the H^- production and have furthermore shown severe erosion due to the high hydrogen particle fluxes. MoLa as well as LaB6, however, both of which are expected to have a work function of slightly below 3 eV, show an increase of the negative ion yield of almost 50 % compared to the reference sample. Further investigations focusing on Cs-free low work function materials are on-going, including measurements of the actual work function under these ion source relevant plasma conditions.

RF-driven ion sources require very high power levels of about 100 kW for each of the eight drivers of the ITER beam source in order to achieve the required ion currents. In order to relax the demands on the RF components and thus increase the reliability of the system, it is highly desirable to improve the RF coupling efficiency. Two alternative concepts are tested in lab-scale: the helicon concept (RF coupling assisted by wave heating) and coupling via a planar ICP antenna enhanced with ferrites. The evaluation of the RF coupling efficiency is based on the quantification of the real part of the RF matching network and plasma coil impedance, which is responsible for the (mainly ohmic) power losses during plasma operation. Consequently, the fraction of the RF power actually coupled to the plasma can be quantified.

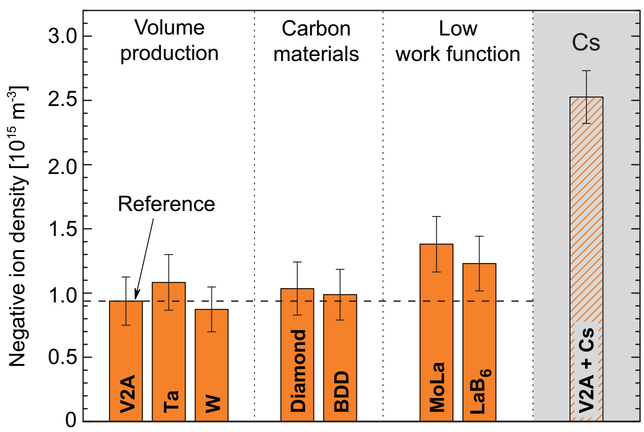

Figure 1: Negative ion densities measured at 2.5 cm distance above surfaces under discussion as potential caesium alternatives.

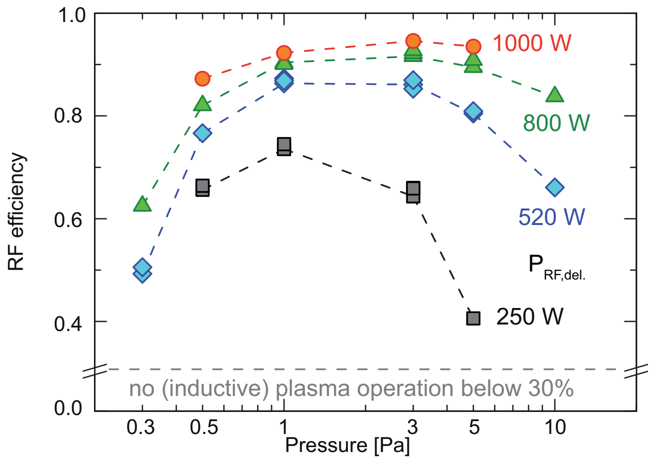

Figure 2: RF efficiency (RF power coupled to the plasma / RF power delivered by the generator) of an ICP in hydrogen at different pressure and power levels delivered by the generator ($P_{RF,del}$).

Figure 2 shows the RF coupling efficiencies obtained at a cylindrical ICP experiment operating at 1 MHz RF frequency for having the reference scenario to ion sources for fusion. With varying pressure, a maximum of the coupling efficiency is obtained between 1 and 3 Pa. The RF efficiency increases with power to 95 % which is attributed to the increased electron density and thus increased plasma conductivity.

Proof-of-principle investigations regarding the feasibility of a laser neutralizer for the neutral beam system for DEMO have been started. The objective of this approach is the realization of a substantial, i.e. measureable, neutralization of a negative ion beam via a photo detachment neutralizer as well as the identification of potential show-stoppers along the way. In a first step, an independent optical system on a low-vibration optical table has been set up, aiming to couple a CW Nd:YAG laser (8 W power at 1064 nm) resonantly to an external high finesse cavity utilizing the Pound-Drever-Hall locking technique. The alignment of the laser beam has been performed according to calculations using ray transfer matrix analysis and experimentally confirmed by measurements via a laser beam profiler. Investigations regarding mode matching and stable locking of the laser frequency on the resonance of the enhancement cavity are ongoing. This will particularly include experiments regarding the stability of the lock when influenced by external disturbances which are present and unavoidable at a neutral beam system, such as mechanical vibrations caused by the gas pumping system.

By the end of 2014, a new collaboration with the team of the Linac4 H^- ion source at CERN has been established. This DFG-funded project is concerned with the improvement of the ion source performance which has only been carried out empirically in the past. For a dedicated optimization, a detailed knowledge of the plasma parameters and the processes taking place in the discharge is mandatory. Therefore, optical emission spectroscopy measurements and a subsequent analysis with the collisional radiative models Yacora H and H_2 are performed.

Low Temperature Plasmas

Hydrogen and nitrogen containing discharges emit intense radiation in a broad wavelength region in the VUV. This emission can attain a remarkable amount of the power used for plasma generation and the corresponding photon fluxes onto surfaces can become relevant for consideration in material processing and surface treatment processes. Consequently, quantification and even control of VUV photon fluxes become desirable. Investigations on this topic have been started using a VUV spectrometer (intensity calibrated between 117 and 280 nm) directly connected to an inductively coupled plasma. Figure 3 shows VUV photon fluxes compared to particle fluxes deduced from the respective densities using emission spectroscopy and Langmuir probes. Clearly, the atomic hydrogen flux dominates, but the photon flux is comparable to the ion flux. Most contributing to the VUV flux is the resonant transition from the first excited state, the Lyman band, followed by the atomic Lyman-α line. Thus, the VUV radiation should not be neglected in considerations of surface modifications. Furthermore, the fraction of the RF power converted to VUV photons is in the order of about 20 %. Hence, it should be considered in power balance models of the discharge.

Ro-vibrational excitation of hydrogen molecules can significantly influence molecular reaction rates in low temperature discharges. The common diagnostic methods such as anti-Stokes Raman scattering (CARS) spectroscopy or VUV laser absorption spectroscopy require complex and expensive laser systems. Therefore, a much easier access to the rotational distribution of the electronic ground state relying on optical emission spectroscopy of the Fulcher-α transition of H_2 ($d\ ^3\Pi_u \rightarrow a\ ^3\Sigma_g^+$) has been successfully developed.

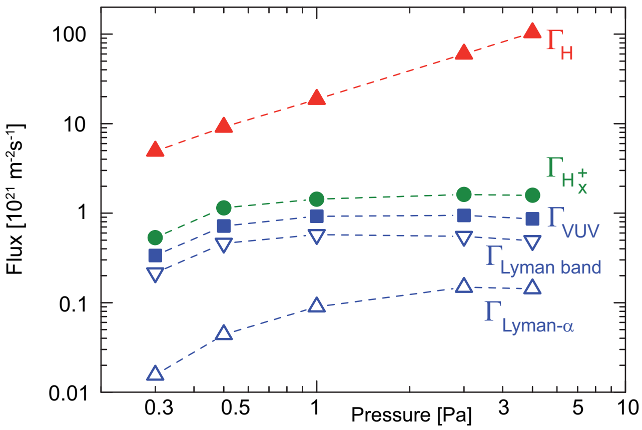

Figure 3: Particles fluxes (atomic hydrogen Γ_H and hydrogen ions Γ_{Hx+} (x=1,2,3)) compared to photon fluxes in the VUV (Γ_{VUV}: 117 – 280 nm) for ICP hydrogen plasmas at an RF power of 540 W. The contributions of the Lyman band (130 – 190 nm) and the Lyman-α line to the total VUV flux are indicated as well.

Theses

M. Müller: Handyspektrometer für den Unterrichtseinsatz. (Thesis for teaching degree program)
D. Schmid: Zeitliche und räumliche Dynamik eines atmosphärischen Plasmoids. (Thesis for teaching degree program)
A. Heiler: Spektroskopische Untersuchungen von Plasmen in einer großflächigen Quelle für negative Wasserstoffionen. (Bachelor Thesis)
M. Siebenhütter: Simulation von Stickstoff Molekülspektren zur Plasmadiagnostik. (Bachelor Thesis)
D. Zielke: Charakterisierung einer Langmuir-Sondendiagnostik an einem induktiv gekoppelten Plasma. (Bachelor Thesis)

Scientific Staff

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Universität Bayreuth
Lehrstuhl für Theoretische Physik V

Head: Prof. Dr. Arthur G. Peeters

The Cooperation between the University of Bayreuth and IPP has focused on research on turbulent transport mechanisms which can produce finite intrinsic toroidal rotation in tokamaks. The local version of the GKW code has been extended to systematically test all the local mechanisms, with the result that, although non-negligible, they are not large enough to reproduce ASDEX Upgrade observations. Global effects are currently under investigation.

In June 2010 the University of Bayreuth opened a new Chair researching the physics of high temperature plasmas. The Chair is financially supported by the University, the 'Volkswagen-Stiftung', through a Lichtenberg Professorship of Prof. A. G. Peeters, and the IPP. Through this Chair the University and the IPP continue and strengthen their long term collaboration, in particular in the areas of nonlinear dynamics and computational physics. The dedication to the collaboration is also expressed through the involvement of an IPP employee, PD Dr. W. Suttrop, in the teaching at the University. The projects deal with toroidal momentum transport, micro-instabilities and turbulence as well as the interaction of small scale turbulence with large scale MHD modes. Below the progress is reported over the last two years (2015 and 2016), where collaborative research has been particularly dedicated to toroidal momentum transport and the physics of intrinsic rotation. The mechanisms which lead to the development of a finite toroidal rotation in tokamak plasmas in the absence of any externally applied torque are still challenging current theory and modelling capabilities. So far only limited attempts have been performed in order to systematically and quantitatively compare the predicted magnitude of the theoretically predicted mechanisms for residual stress and finite intrinsic rotation with the experimental measurements. Within the collaboration between IPP and the University of Bayreuth, a research project has been defined which: 1) extends the gyrokinetic code GKW to include all of the relevant mechanisms of residual stress in the confined plasma domain 2) systematically applies the GKW code to a large set of experimental data to identify the dominant ones and test whether the predicted size and parametric dependences can explain some key observations, like the intrinsic rotation reversal with increasing density in Ohmic plasmas. A substantial part of the activity in 2015 has been dedicated to the development and the benchmarking of the coupling between the neoclassical drift-kinetic code NEO and the gyrokinetic code GKW. This coupling allows the calculation of the impact of the distribution function produced by the neoclassical equilibrium on the turbulent transport. The explicit dependence of the neoclassical corrections on the parallel velocity (in contrast to the usual Maxwellian distribution) introduces a symmetry breaking mechanism which produces a residual stress. Successful benchmarking has been performed with a similar previous implementation in the gyrokinetic code GS2. The application to a large set of observations in ASDEX Upgrade revealed that this mechanism produces non-negligible levels of residual stress, comparable to the momentum fluxes produced by other mechanisms like the Coriolis pinch and the up-down asymmetry of the magnetic flux surfaces. However none of these mechanisms is able to produce levels of residual stresses which are consistent with the size of intrinsic toroidal rotation gradients which are experimentally measured, and which can be up to a factor of 5 larger. In addition, residual stresses produced by higher order parallel derivatives in the gyrokinetic equation are negligible, at least in the considered range of experimental conditions. From this set of results obtained within the local limit, the project is currently moved to exploring the effects connected with the symmetry breaking produced by radial profile variations, which can be tested only by means of a global code. The global version of GKW has been extended to also treat shaped geometry and has been interfaced to ASDEX Upgrade experimental profiles. Moreover, specific diagnostics for the study of toroidal momentum transport with global simulations have been added. Global simulations of experimental cases, as well as global benchmarks, are currently carried out to test the impact of kinetic electron physics on the dependence of the residual stress on the normalized Larmor radius, the impact of the turbulence type. A parallel effort is dedicated to the study of the impact of the neoclassical background symmetry breaking on the transport of impurities, which is particularly affected in the presence of a radial gradient of the toroidal velocity. It is found that this effect can become non-negligible for heavy impurities at high collisionalities, like those achieved at the edge of present tokamaks.

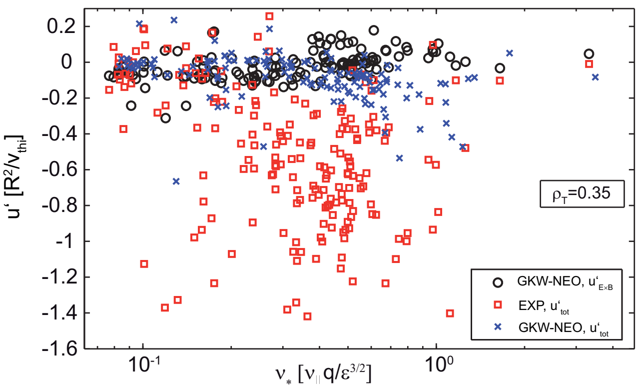

Figure: Radial gradient of the toroidal flow predicted by the GKW and NEO coupling (E×B flow only is plotted with black cyrcles, total flow with blue crosses) as a function of the ion-ion collision frequency and compared to the experimental measurements (red squares), at the radial position $\rho = 0.35$.

Scientific Staff

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Technische Universität Berlin
Plasmaphysik, Plasma-Astrophysik

Heads: Prof. Dr. Robert Wolf, Prof. Dr. Wolf-Christian Müller

As a cooperative effort of TU Berlin and IPP, the groups carry out interdisciplinary research and university tuition incorporating the physics of high-energy and laboratory plasmas as well as nonlinear plasma dynamics. The Plasma-Astrophysics group is part of the International Max-Planck-Princeton Research Center for Plasma Physics (MPPC) and of the Berlin International Graduate School in Model and Simulation based Research (BIMoS).

Lagrangian Statistics of Turbulence

The transport properties of turbulent media are most naturally analyzed in the co-moving Lagrangian frame of reference. The Lagrangian theory of turbulence has consequently experienced a tremendous growth of importance, knowledge, and impact over the last decade. The pioneering activities by some members and collaborators of the plasma-astrophysics group have led to an international scientific collaboration involving this group and researchers based in the United Kingdom, in particular at the universities of Glasgow, Exeter and Warwick (Centre for Fusion, Space, and Astrophysics). The efforts of this activity continue. Initial studies on the basic properties of the convex hull of many Lagrangian tracer particles passively advected by turbulent flows have been completed. Geometrical measures quantifying differences in the evolution of the hull and the enclosed particle ensemble have been shown to reliably indicate different turbulent transport regimes. Studies on the utility of this concept for detection and characterization of flow anisotropy in the transition toward two-dimensional dynamics in plasmas subject to a mean magnetic field of increasing amplitude have been started. This project advances in parallel to the investigation of MHD turbulence in mean magnetic fields in the Eulerian frame of reference, described below.

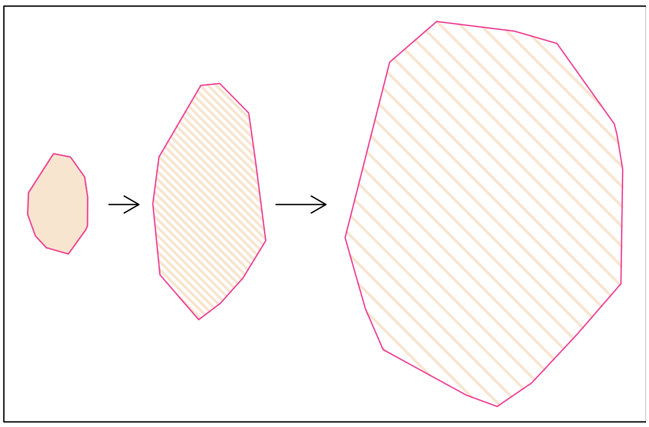

Figure 1: Evolution of two-dimensional convex hull surrounding group of dispersing particles. Particles shown in each step are the same.

Theoretical efforts include the application of extreme value theory to convex hull properties and the numerical verification of random walk models applied to convex hull evolution in two-dimensional turbulence.

MHD Turbulence in Strong Magnetic Fields

The work on magnetohydrodynamic turbulence in a strong mean magnetic field, B_0, has been continued. Based on the application of Yaglom's 4/3-law to anisotropic turbulence a direction-specific representation of nonlinear cascade fluxes has been obtained and successfully tested on different turbulent flows. Of particular interest is the new regime of weakened MHD turbulence (isotropic scaling w.r.t. the mean field direction, scale-independent fluctuation anisotropy $\sim b_{rms}/B_0$, significant field-parallel transport). It develops under certain conditions and represents a bridge between weak turbulence $\tau_{nonlinear} >> \tau_{linear}$ and strong turbulence $\tau_{nonlinear} \sim \tau_{linear}$, the τ symbols denoting characteristic timescales.

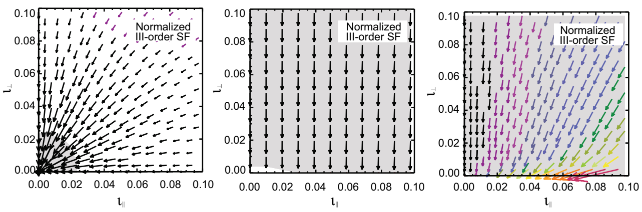

Figure 2: Representation of energy flux in MHD turbulence w.r.t. spatial scale parallel and perpendicular to mean magnetic field $\boldsymbol{B}_0$. Left: globally isotropic strong regime B_0=0, middle: anisotropic strong regime B_0=5. Right: weakened regime B_0=5 with non-negligible parallel flux component (coloured).

This regime exhibits a clear dependence on the largest-scales of the flow which drive the turbulence.

Modelling ECRH Plasma Start-up in Wendelstein 7-X

The plasma start-up driven by electron cyclotron radiation heating (ECRH) in a stellarator device is of natural importance for the operation of the Wendelstein 7-X experiment. A Diploma thesis carried out at TU Berlin under the joint supervision of R. Wolf and W.-C. Müller on the quantitative numerical modelling of this process, has resulted in the successful implementation and validation of a simple rate-equation model for the ECRH-driven start-up of a circular hydrogen plasma. This work has shown that the simplifying assumption of diffusive spatial transport made to introduce an approximate spatial dependence into the model requires further refinement. It has triggered the development of a true three-dimensional multi-fluid simulation model for further studies on this subject.

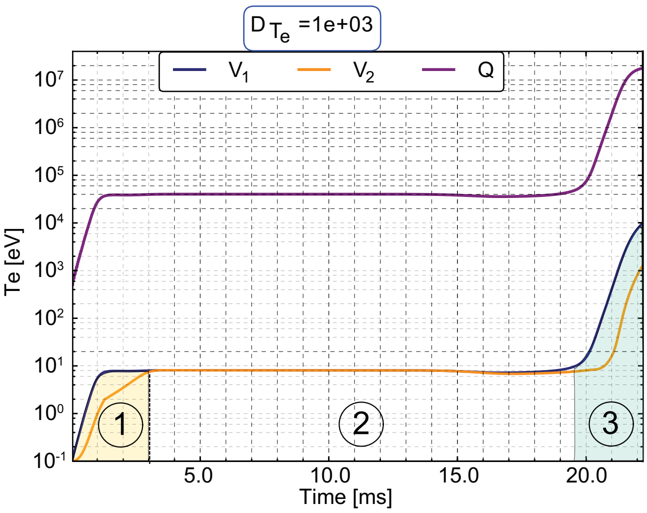

Figure 3: Evolution of electron temperature, T_e, and heating rate, Q, for the sub-volume subject to direct microwave irradiation V_1 and the rest of the plasma vessel V_2. Following an initial ramp-up phase (1) and a plateau phase characterized by increasing ionization levels (2), the fully-ionized plasma state is reached after about 20 ms.

Topology-driven Magnetic Reconnection

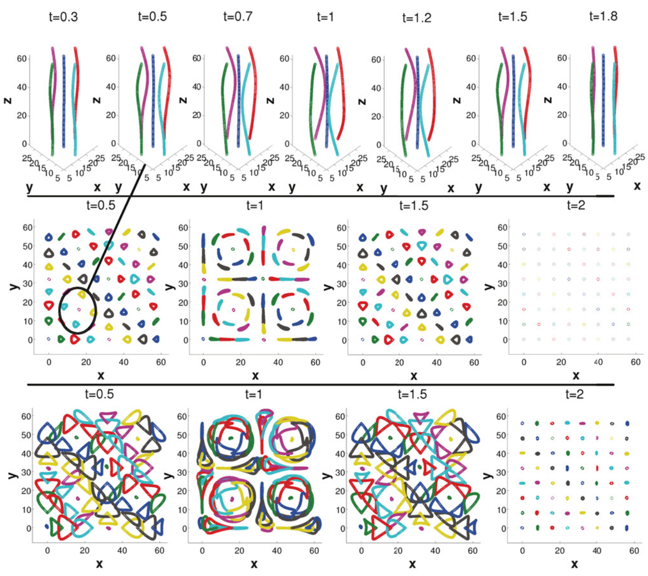

Figure 4: Example of magnetic field lines viewed along the vertical for two different forcing amplitudes F ~ 0.1 (second row), F ~ 0.3 (third row). The first row gives a different perspective on the lines of force for F ~ 0.1.

Motivated by a recent model of magnetic reconnection driven by the exponential separation of neighbouring magnetic field-lines, a numerical experiment of a plasma confined between two horizontal, highly conducting boundaries that is permeated by a vertical mean magnetic field has been implemented in the course of an international double-master thesis. The boundary conditions ensure that the field lines are fixed at the positions where they leave the horizontal plates limiting the plasma vertically. A solenoidal force field is driving the plasma velocity, thereby deforming the magnetic field lines with the aim of increasing field-line exponentiation which should eventually result in magnetic reconnection. The spectral model does not suffer from the numerical diffusion present in classical reduced MHD codes which have already been applied to the problem. This numerical system will be employed to investigate the details of this possible form of reconnection in the future.

Nonlinear Interactions and Detailed Energy Fluxes in MHD Turbulence

The detailed nonlinear terms in the simple model system of incompressible magnetohydrodynamics exhibit a broken symmetry due to the additional ideal conservation of cross-helicity in this system.
This allows for the solution of the classical turbulence problem of determining the detailed three-mode nonlinear energy exchange. In the framework of this project a numerical tool is being developed to analyze simulation data with the aim of obtaining a detailed picture of the multi-mode energy exchange. The results will allow to create better reduced models of turbulence beyond the shell-model paradigm and to gain deeper understanding in the fundamental nonlinear dynamics of turbulence.

The Shock-capturing MHD Simulation Code *csmhd*

In the context of the magneto-rotational-instability-project within the Max-Planck-Princeton Center of Plasma Physics and a project on the Lagrangian statistics of supersonic turbulence (see below), a high-order finite-volume shock-capturing solver for the hyperbolic conservation laws governing fluid turbulence with and without magnetic fields has been developed and successfully tested. It employs a constrained-transport scheme to enforce the solenoidality of the magnetic field and combines fourth-order accuracy with the simple and efficient local Lax-Friedrichs flux approximation. The code will be applied to problems in supersonic flows.

Lagrangian Properties of Supersonic Turbulence

In contrast to turbulence in incompressible fluids, the supersonic turbulent flows often encountered in astrophysical settings exhibit significant density variations which are reflected by clustering of the passive tracer particles tracked in Lagrangian turbulence investigations. This project aims at a comprehensive characterization and theoretical description of density fluctuations by use of Lagrangian statistics and the exploitation of the spatial locality the convex hull of compressively induced tracer particle clusters. The simulation code *csmhd* has been extended by a parallelized particle tracking module which will be utilized for this purpose in the investigation of supersonic turbulence simulations.

Ernst-Moritz-Arndt-Univ. Greifswald – Technische Univ. München

APEX, PAX, NEPOMUC

Head: Prof. Dr. Thomas Sunn Pedersen

A pair plasma, consisting of an equal amount of electrons and positrons, is of great interest for fundamental plasma physics, as well as for astrophysics, where such plasmas are believed to exist near very energetic astrophysical objects. In the APEX project (A Positron-Electron EXperiment), we aim to create the first confined electron-positron plasma in a laboratory. Initial experiments on positron trapping in a novel geometry were successful.

Overview

The two main challenges in the creation of a confined positron-electron plasma are the design and construction of a suitable (contact-free, low-loss) storage configuration, and the command over a large amount of positrons, which are notoriously difficult to produce in large quantities on Earth. To address the second challenge, APEX is being prepared at IPP-Garching, and will be installed at the NEPOMUC beamline of the FRM II research reactor, also located in Garching. NEPOMUC delivers the highest flux of cold positrons that is currently achievable world-wide. Techniques for massive positron accumulation are investigated in Greifswald in the Positron Accumulation EXperiment (PAX). The whole project strongly benefits from its national and international collaborations with the University of Greifswald (Lutz Schweikhard), UC San Diego, CA (Cliff Surko), Lawrence University, Appleton, WI (Matt Stoneking) and with the National Institute of Fusion Science (NIFS), Toki, Japan (Nagato Yanagi).

APEX

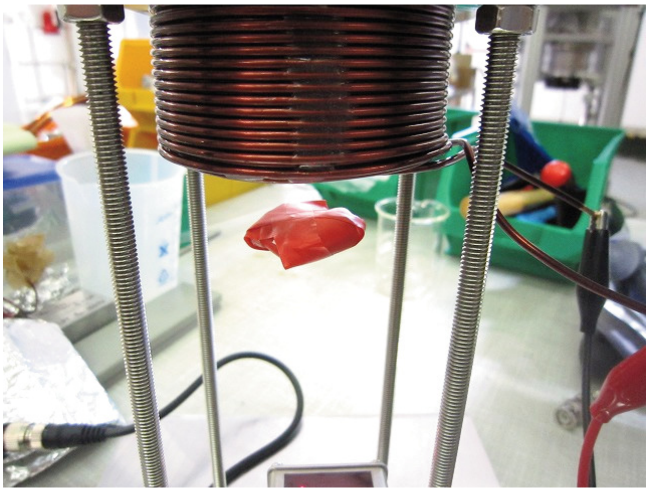

Figure 1: Test device for levitation techniques. A small permanent magnet (wrapped in red adhesive tape) is levitated by a field coil. Levitation is actively stabilized by monitoring the magnet position with a laser sensor (below), which controls a PID circuit.

The device for confining the electron-positron plasma must be capable of trapping very low density plasma, using only external fields, and potentially at varying degrees of neutralization. Two magnetic field configurations suitable for this purpose have been identified: A stellarator, and, more recently, a magnetic dipole field created by a levitated, superconducting current loop. Both have been and will be considered, but we are currently focusing on the levitated magnetic dipole (current ring), partly because of the astrophysical relevance of this topology. The levitation, inside the vacuum chamber, is achieved by the feedback-controlled attraction to a second electromagnet placed above the dipole magnet.

Stable feedback-controlled levitation has been achieved in our laboratory. For this test, a permanent magnet was used as the object being levitated (figure 1). The same technique will be used to levitate the coil that will produce the confining magnetic field in the APEX device. A suitable coil, made of a high-temperature superconductor, currently is under construction in a joint project with NIFS in Japan.

Our strategy for producing a confined electron-positron plasma relies on the use of an external positron source. From there, positrons need to be transported perpendicular to the magnetic field into the confinement region of APEX. Several different strategies have the promise to solve this problem, among which we have identified drift injection by using a pair of E×B plates as the one that can be most easily implemented on a short time scale.

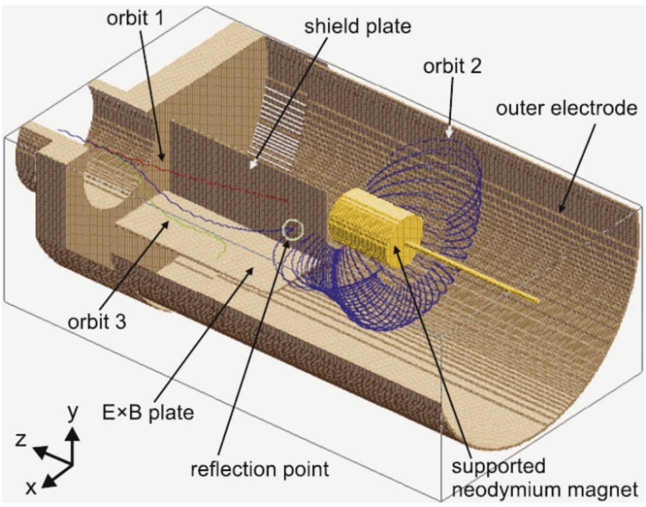

Figure 2: Simulation of positron orbits in a prototype dipole field trap. View from below and to the side of the magnet. Before approaching the magnetic field of a rare-earth permanent magnet (center), particles (entering from the top left) transit through a pair of parallel 'E×B' plates (only one of them shown). Typical trajectories (orbit 2, in blue) undergo a drift motion transporting them into stronger regions of the B-field. Without the E×B plates, all particles would be reflected back when attempting to enter into the magnetic field (orbit 1).

Starting in December 2014, we carried out several experimental campaigns at NEPOMUC, in which we have tested the efficiency of positron injection into a magnetic dipole field created by a strong permanent magnet. By using the E×B drift between a pair of parallel plates sideways of the magnet, we can modify the trajectories of a large fraction of the positrons such that they end up precessing around the permanent magnet in the equatorial region (figure 2). With a slightly refined version of the field configuration in the figure, we have recently achieved an injection efficiency of 90 % for slow positrons.

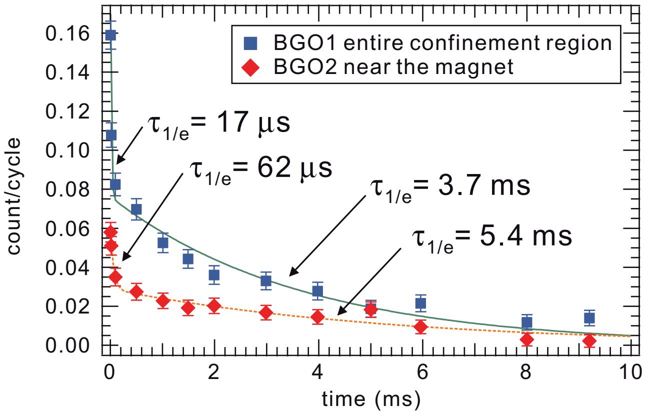

Figure 3: Positron annihilation events measured with a scintillation detector, after filling the confinement region of a permanent magnet by the method shown in figure 2. The positron beam was electrostatically blocked at t=0, such that the lifetime of the trap content could be monitored. The two signals are from detectors monitoring different regions within the trap. A bi-exponential decay with a longer component in the ms range is clearly seen.

Positron confinement times have been measured in the same experiment by pulsing the injected positron beam and subsequently measuring the decay time of the positron annihilation signal (figure 3). We have thus achieved lifetimes in the ms range.

NEPOMUC Beam Characterization

In close cooperation with the NEPOMUC team at FRM II (Christoph Hugenschmidt et al.), we have quantitatively characterized the properties of the positron beam at the open beamport, which is the beamline available for general user experiments. Two different variants of the positron beam can be used: Either a high kinetic energy beam directly transported from the source region, or a lower intensity, but higher brilliance beam produced by scattering of the primary beam on a W crystal ('remoderation'). Both beams differ in intensity, monochromaticity, and ratio of the parallel ($E_\parallel$) to perpendicular ($E_\perp$) energy, quantities that are all important to our experiments. An accurate knowledge of these parameters is important for optimizing the injection and accumulation of positrons for our purposes. The absolute positron flux was between 2-6×10^7 for the remoderated beam (5-22 eV) and at 5.4×10^8 for the direct beam (400 eV), further results are tabulated below. These measurements used a residual field analyzer constructed at IPP.

Energy (eV)	$\Delta(E_\parallel)$ (eV)	$\langle E_\perp \rangle$ (eV)
5	1.8(1)	0.78(3)
12	2.5(2)	1.22(1)
22	3.1(2)	1.3(1)
400	~40	~25

PAX

A substantial part of work in the PAX project is done using electrons as proxies for positrons. We have therefore investigated whether widespread charged particle detection schemes can be relied on for comparing the two species. For a commonly used luminescing material ('P20') we found substantial differences in the sensitivity for positron compared to electron detection (figure 4); most remarkably, even for very low positron energies, the luminescence yield does not approach zero. This is, to the best of our knowledge, described here for the first time.

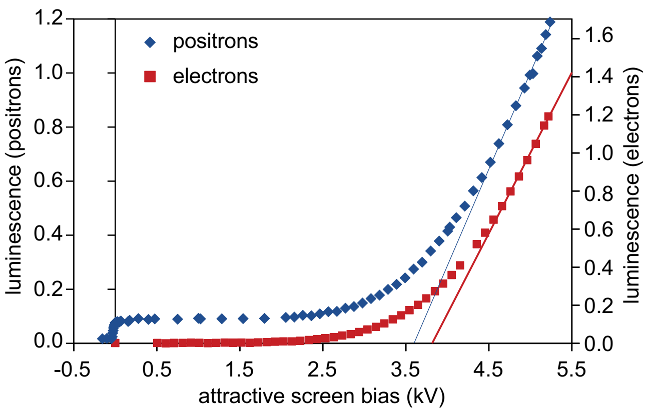

Figure 4: Luminescence efficiency of positron (left axis) and electron currents (right axis) of comparable strength on a commercial fluorescence screen, as a function of screen bias. Clearly positrons lead to more efficient luminescence, in particular at low screen bias. Both curves have independently been normalized to the luminescence intensity at 5 kV screen bias. At this energy, luminescence due to positrons is 1.42(8) times as intense as the one from electrons.

Scientific Staff

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Universität Stuttgart

Institut für Grenzflächenverfahrenstechnik und Plasmatechnologie (IGVP)

Head: Apl. Prof. Günter Tovar

The joint program between IGVP and IPP on ECRH systems for AUG, W7-X, and ITER as well as contributions to the experimental program of AUG can be found on the respective pages of this report. Here is summarized the part of the program carried out at IGVP: the development of new mm-wave components, investigations of plasma waves and turbulent transport. Experiments are carried out on the torsatron TJ-K, which is operated with a magnetically confined low-temperature plasma.

Microwave-Plasma Interaction

The interaction between electromagnetic waves and the plasma is one of the research topics at the stellarator TJ-K. This includes heating by externally injected microwaves as well as waves emitted by the plasma itself. Experiments are complemented by simulations with the cold plasma full-wave code IPF-FDMC.

TJ-K is currently equipped with three different microwave heating systems, operating at 2.45 GHz, at 7.9-8.4 GHz, and at 13.75-14.5 GHz. The power of the 14 GHz system has been upgraded to an output power of 6 kW by installing a third klystron with a separate transmission line. A phased array antenna allows to sweep the injection angle without the necessity to mechanically move some parts. With increased heating power, an increase in the plasma density is observed. At low neutral gas pressure, however, indications of a regime with reduced collisionality are found.

In order to estimate energy confinement times in TJ-K, the potential of two diagnostics measuring the radiation emitted by the plasma has been explored. An optical diode, detecting radiation in the visible range and a commercial satellite receiver (LNB) have been used. While for the radiation in the visible range agreement with values obtained from a particle and energy balance model could be achieved, the radiation detected by the LNB can be due to cyclotron radiation and/or bremstrahlung.

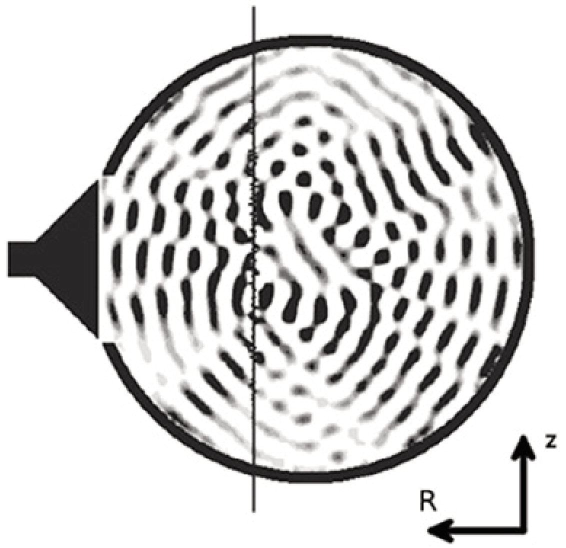

Figure 1: Electric-field amplitude distribution of microwaves emitted from sources, located at the vertical line, into the poloidal torus cross-section. A receiver antenna is positioned at the outboard midplane.

To better understand the microwave radiation emitted by the plasma in TJ-K, simulations with the full-wave code have been performed, resembling radiation measurement at the second harmonic of the electron cyclotron resonance. As can be seen in figure 1, the field distribution at the receiving antenna is strongly influenced by reflections from the metal wall. Density fluctuations will be included as a next step to study their influence on the microwave propagation.

Global Turbulence and Confinement Studies

In fusion devices, an increased zonal flow activity is found when the bifurcation to high confinement (H-mode) is reached. With their m=n=0 topology and finite radial wave number k_r, the zonal flow is intrinsically connected to a poloidal shear flow. Drift-wave eddies are tilted and drive the shear flow, which leads to a self-amplification of the zonal flow. A key parameter in the drift wave - zonal flow system is the collisionality C. It determines the coupling strength between the density and potential. Using a poloidal probe array various measurements were carried out in the low-temperature plasmas of the stellarator TJ-K, to investigate the collisionality scaling of zonal flows. In figure 2, the relative zonal flow power P_{ZF}/P_{total} as a function of the collisionality is shown. In the limit of the adiabatic case $C \rightarrow 0$, the zonal flow contribution to the complete spectrum strongly increases. For low collisionality, density and potential are strongly coupled and act similar. Since the zonal flow is a pure potential structure, but the drift waves are sheared in the density, the drive is more efficient for higher coupling.

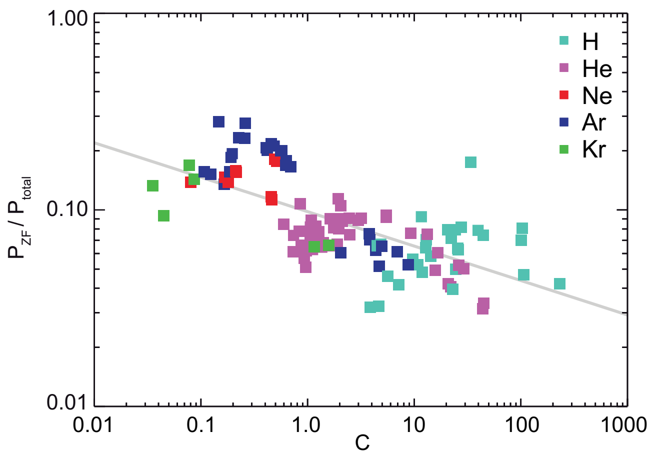

Figure 2: Collisionality scaling of relative spectral zonal flow power for various gases. The zonal flow power increases with decreasing collisionality.

In the hydrodynamic limit, on the other hand, the potential decouples from the density. As a result density is merely advected passively. Numerical simulations of the Hasegawa-Wakatani model suggest that in the hydrodynamic limit, density fluctuations become more intermittent. This has been confirmed for density fluctuations measured in the confinement region of TJ-K, whereas the collisionality was varied mainly through working gas type and pressure.
Furthermore, the 3D structure of transient filamentary high-amplitude events in the scrape-off layer (SOL), so-called blobs, has been analysed with respect to its dependence on the magnetic field structure, especially magnetic shear. To this end, 2D fluctuation measurements were performed at two different toroidal positions, simultaneously. The perpendicular shape of blobs measured at one position was numerically mapped along field lines onto the second toroidal position and compared to the measured shape there. In this way, the field alignment of blob filaments was traced in time along the perpendicular blob trajectory. Blobs were found to be field-aligned close to the last closed flux surface, but become misaligned further into the SOL. In fact, as they propagate the filaments do not deform due to the magnetic shear, rather they retain their original form and propagate in a rigid manner.

Doppler Reflectometry Simulations with IPF-FD3D

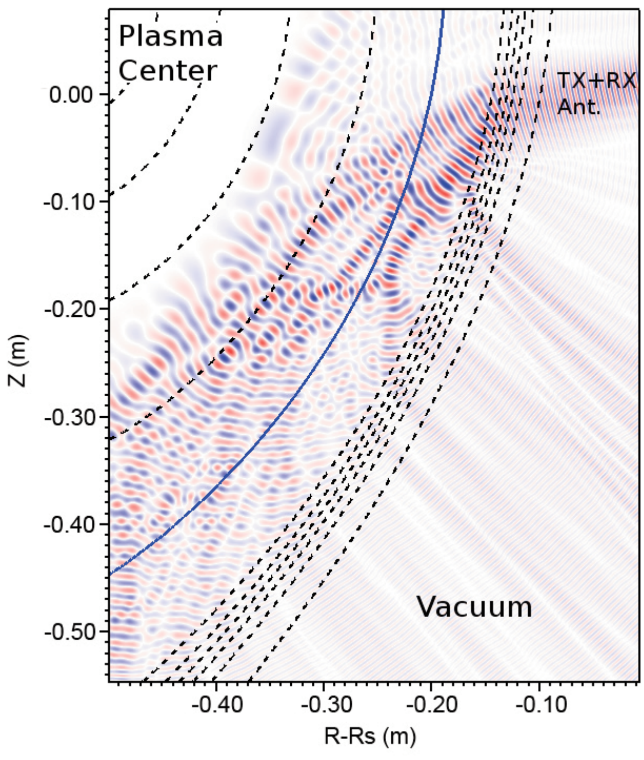

Figure 3: Wave field of the microwave with overlaid density contours of the plasma. The solid blue line is the cutoff density. The antenna is in the upper right corner.

Doppler reflectometry is an important microwave diagnostic for turbulent fusion plasmas. The incoming wave is scattered at certain density fluctuation wavenumbers, depending on frequency and angle between beam and density gradient. This way, a wavenumber resolved density fluctation spectrum can be measured. However, the correspondence between fluctuation power and scattered microwave power is strongly non-linear. The role of simulations is two-fold: the plasma turbulence code GENE is used to simulate the plasma fluctuations, and the fullwave code IPF-FD3D is used to simulate the Doppler reflectometer with the turbulence field given as input. Figure 3 shows the microwave field propagating in a turbulent plasma. The resultant spectra are shown in figure 4. Turbulence spectra are characterised by the power law decay at large k. Both the experimental and the simulated (A=1) spectra have very similar exponents. If the input turbulence amplitude is decreased by a factor of 10 (A=0.1), the spectral shape changes significantly. This shows the non-linear saturation that affects the A=1 spectrum.

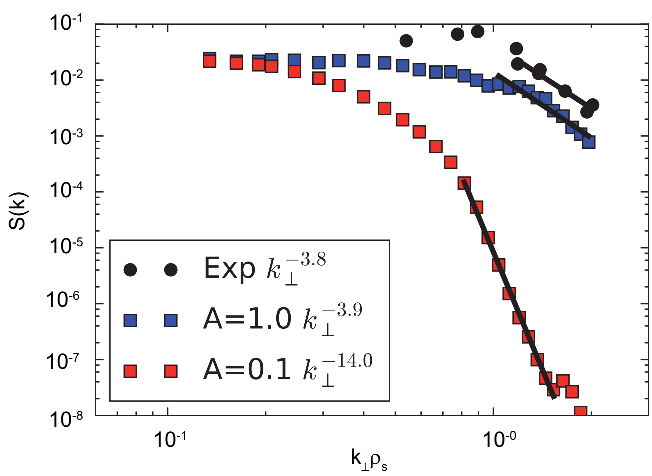

Figure 4: Simulated wavenumber spectra and comparison to experimental measurements. "A" is the scaling of the turbulent fluctuations.

This work was partly performed in the framework of the Helmholtz Virtual Institute on Plasma Dynamical Processes and Turbulence Studies using Advanced Microwave Diagnostics. The fullwave simulations were performed on hazelhen of the High Performance Computing Centre Stuttgart (HLRS).

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Universities in Belgium and France
IShTAR

Heads: Dr. Kristel Crombé, Dr. Rodolphe D'Inca, Prof. Dr. Jean-Marie Noterdaeme

IShTAR (Ion Sheath Test Arrangement) is a linear magnetic facility developed by and operated at the IPP Garching in cooperation between IPP Garching, several universities and institutes. Its purpose is to reproduce conditions representative of a tokamak edge to investigate the interactions between the ICRF antenna and the plasma, especially the rectified electric field produced on the wall, which is the source of spurious effects (sputtering, hot spots, loss of coupling).

Introduction

Several universities and institutes contribute substantially to IShTAR: a satellite experiment to probe the electric field of ICRF antennas.

The Ion Sheath Test ARrangement (IShTAR) is a linear magnetic device equipped with an external Radio-Frequency plasma source able to produce conditions representative of a tokamak's edge to analyze the interactions between the ICRF antenna and the plasma, especially, the creation of a rectified electric field in the sheath near the walls that modifies the plasma transport and induces spurious effects like sputtering, hot spots and degraded power coupling. It supports ICRF operations on ASDEX Upgrade and the developments for ITER by providing flexible control of the experimental parameters, dedicated time, easy access for diagnostics and a fast turn-around time for modifications.

Partners

IShTAR was selected as MST2 project and is funded in part by EUROfusion. It is a cooperation between IPP Garching which assembles and operates the machine and several international partners. UGent provides the MST2 project leader, supervises the physics integration and delivers hardware (Langmuir probes and a powerful spectroscope funded by subsidies from the Federal Ministry of Economy, Belgium). LPP-ERM/KMS (Brussels, Belgium) provides theoretical support and hardware (ICRF antenna). The University of Lorraine (Nancy, France) supports the work on diagnostics with specialist manpower. The University of Aix-Marseille (France) and ORNL (Oak Ridge, USA) are actively following the development of tools to measure the electric field near the antenna.

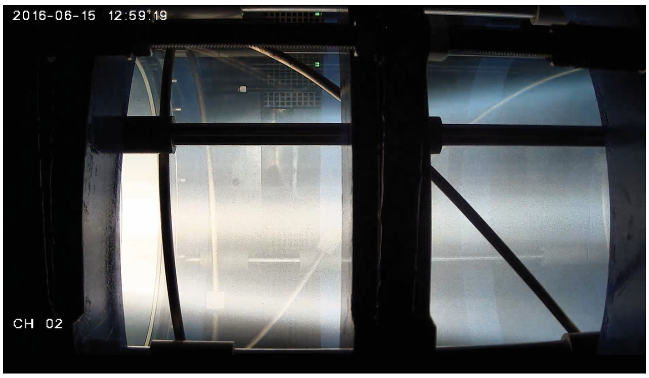

Figure 1: Plasma flow in the RF source.

Ongoing Activities

The activities in 2016 are split in three parts:

- to reproduce an operational environment representative of the tokamak antenna region: a 1×1 m cylindrical vacuum vessel is equipped with coils to produce a 0.2 T linear magnetic configuration; the plasma (figure 1) is produced by an external helicon antenna operating at 3 kW, 0.1 T and Helium or Argon (Hydrogen possible) (figure 2).

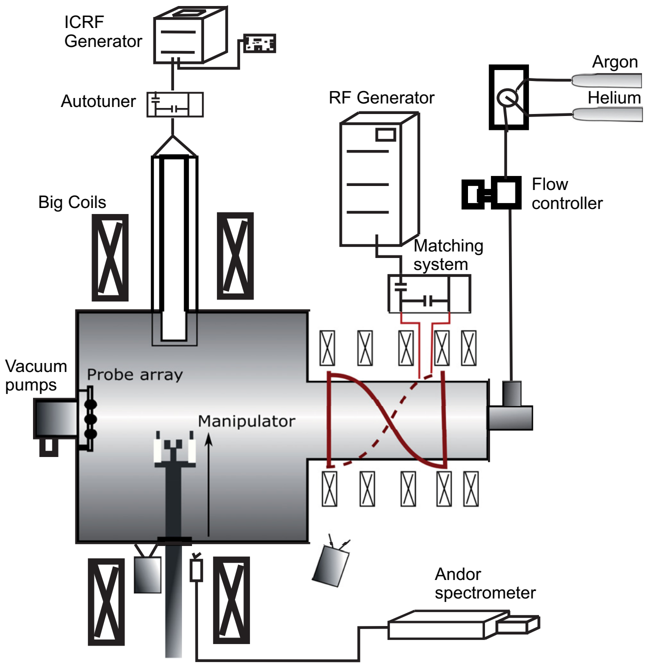

Figure 2: Testbed configuration.

The plasma parameters (temperature and density profile) are controlled through the neutral gas pressure (10^{-5} mbar), the injected RF power and the magnetic field. The purpose is to have a plasma where the relevant plasma waves (the Fast and Slow wave) are distinguishable. For this purpose, the characteristics of the plasma are measured with Langmuir probes (radial density and temperature profile) and with B-dot probes (radial profile of Radio-Frequency components of the field) (see figure 3).

- to design and test reduced-scale ICRF antennas: for the moment, a simple single strap antenna without Faraday Shield (see figure 4) is available and integrated on the testbed: the challenge here is to get an antenna both representative of an operational ICRF system and compatible with the testbed specificities (small plasma radius, creation of eigenmodes).

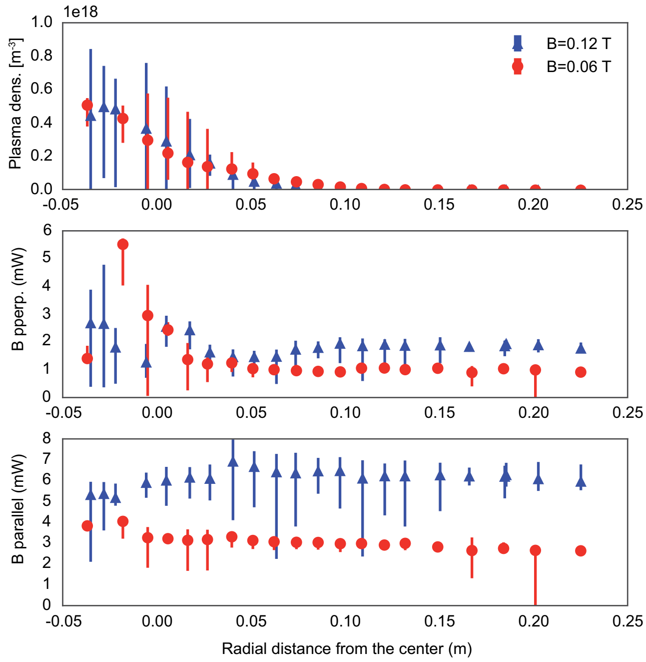

Figure 3: Radial profile of plasma density and perpendicular and parallel RF B-field.

- to develop the techniques to measure both the DC and RF electric field in the very narrow region of the plasma sheath on the antenna walls. Several methods are investigated, most of them based on the Stark effect. The challenge is to extract a useful signal from the background noise created by the helicon source, the static magnetic field and the low gas pressure. As a first step, a high resolution spectrometer with iCCD camera is operated to measure the signal on test electrodes powered by DC voltage up to 20kV located in the plasma source (maximum pressure and density): we validate the method and estimate its limitations in terms of maximum electric field measurable.

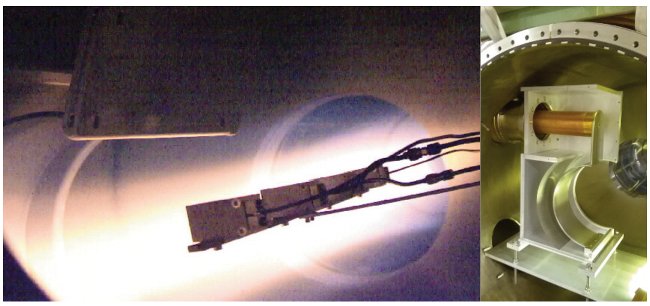

Figure 4: ICRF antenna in plasma with B-dot probes and internal structure.

The aim is to provide data to benchmark the sheath modelling codes and paves the way for the development of reliable ICRF systems on fusion devices.

Contribution to Training and Education

Two master students (one in 2014, one in 12015) did the work for their master thesis on IShTAR. In 2015 two PhD students started their PhD research (one funded by an Erasmus Mundus Fusion-DC fellowship, one funded by a FWO project – Fonds voor Wetenschappelijk Onderzoek, Belgium).

Conclusion

IShTAR is thus designed as an experimental platform dedicated to international cooperations: to improve remote access for participants, it is equipped with an intranet to manage the operations and the documentation and a processing system with sharable notebooks and access to the discharge database.
With these ongoing activities, IShTAR provides a complete experimental platform for students and international collaborators who need not only to investigate phenomena between the ICRF antenna and the plasma but also to validate their codes with data in simplified geometries or their diagnostic in flexible conditions before implementation on an operational tokamak.

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Teams

ASDEX Upgrade Team

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W7-X NBI Team

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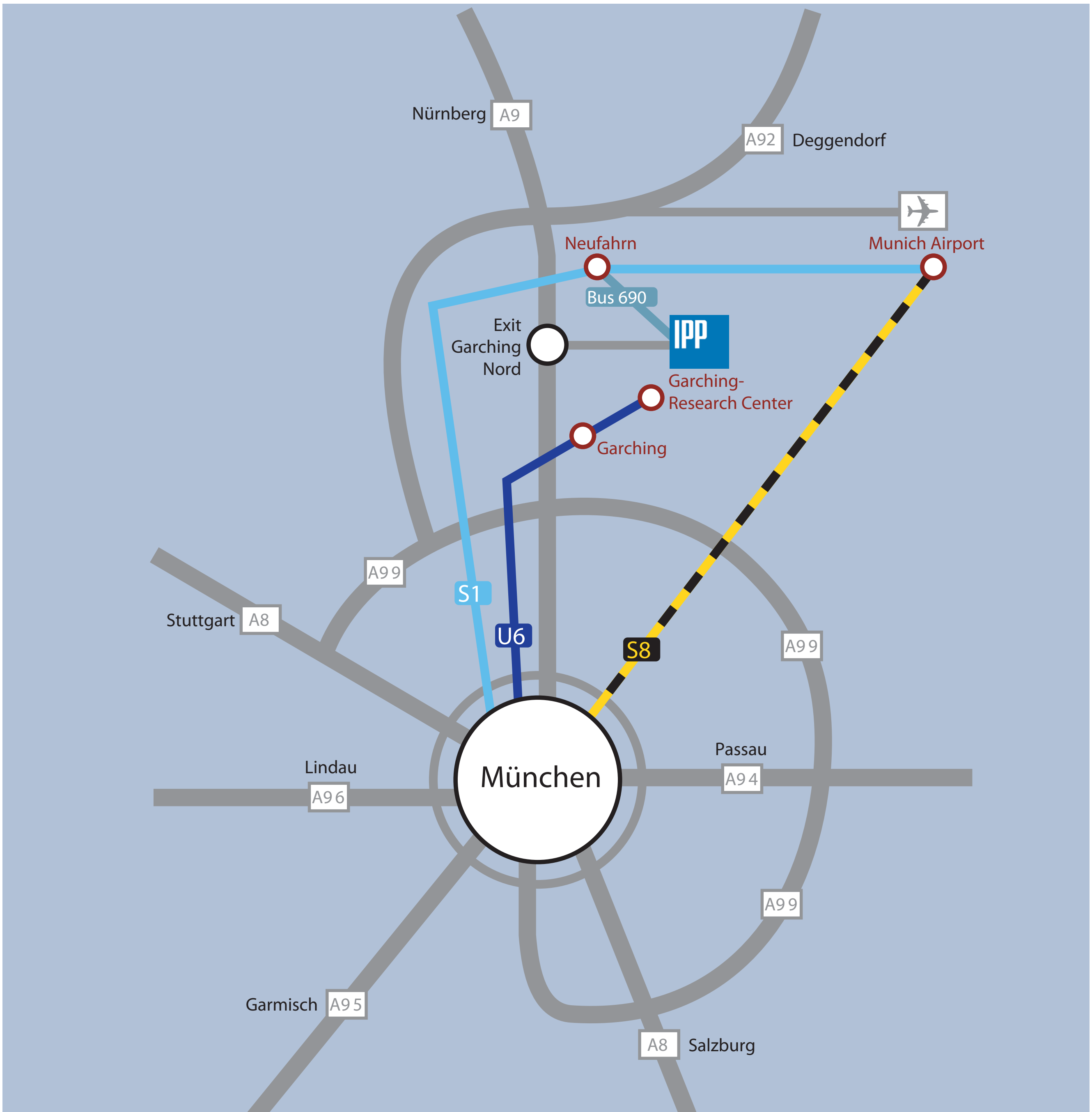
Appendix

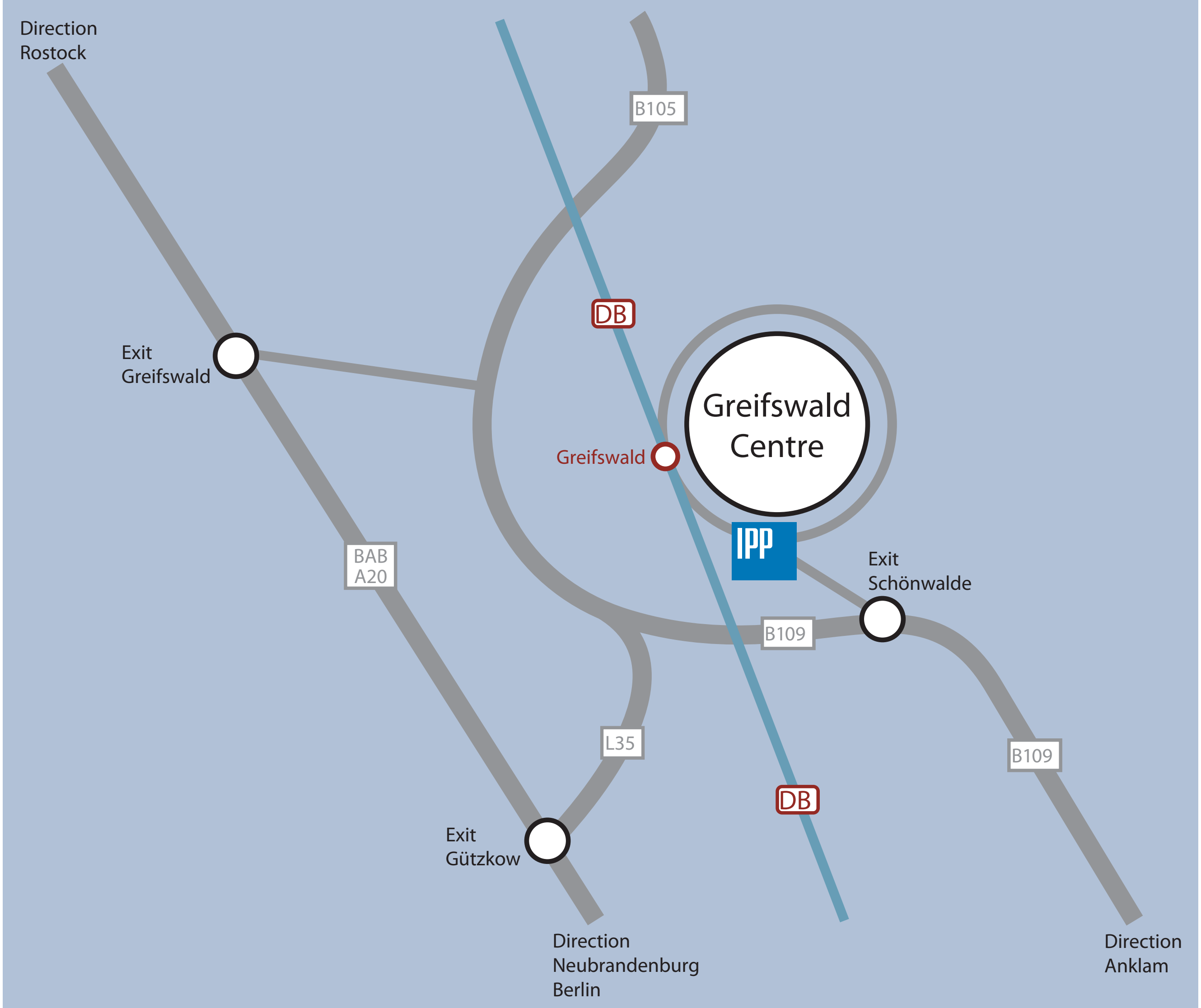
How to reach IPP in Garching



By car:
Exit Garching-Nord on the
Autobahn A9 München-Nürnberg,
then follow the signs “Forschungsinstitute”.

By public transport:
Any S metro from Munich Main Station to Marienplatz,
metro U6 to Garching-Forschungszentrum;
or from Airport Munich: S1 to Neufahrn, then bus 690
to "Garching Forschungszentrum" (only on weekdays).

How to reach Greifswald Branch Institute of IPP



By air and train:
Via Berlin: from Berlin Tegel Airport by bus “JetExpressBus” to Hauptbahnhof (central station), by train to Greifswald.
Via Hamburg: from the airport to main Railway Station, by train to Greifswald main station.

By bus:
From Greifswald-Railway Station (ZOB) by bus No. 3 to the "Elisenpark" stop.

By car:
Via Berlin, Neubrandenburg to Greifswald **or** via Hamburg, Lübeck, Stralsund to Greifswald, in Greifswald follow the signs “Max-Planck-Institut”.

IPP in Figures

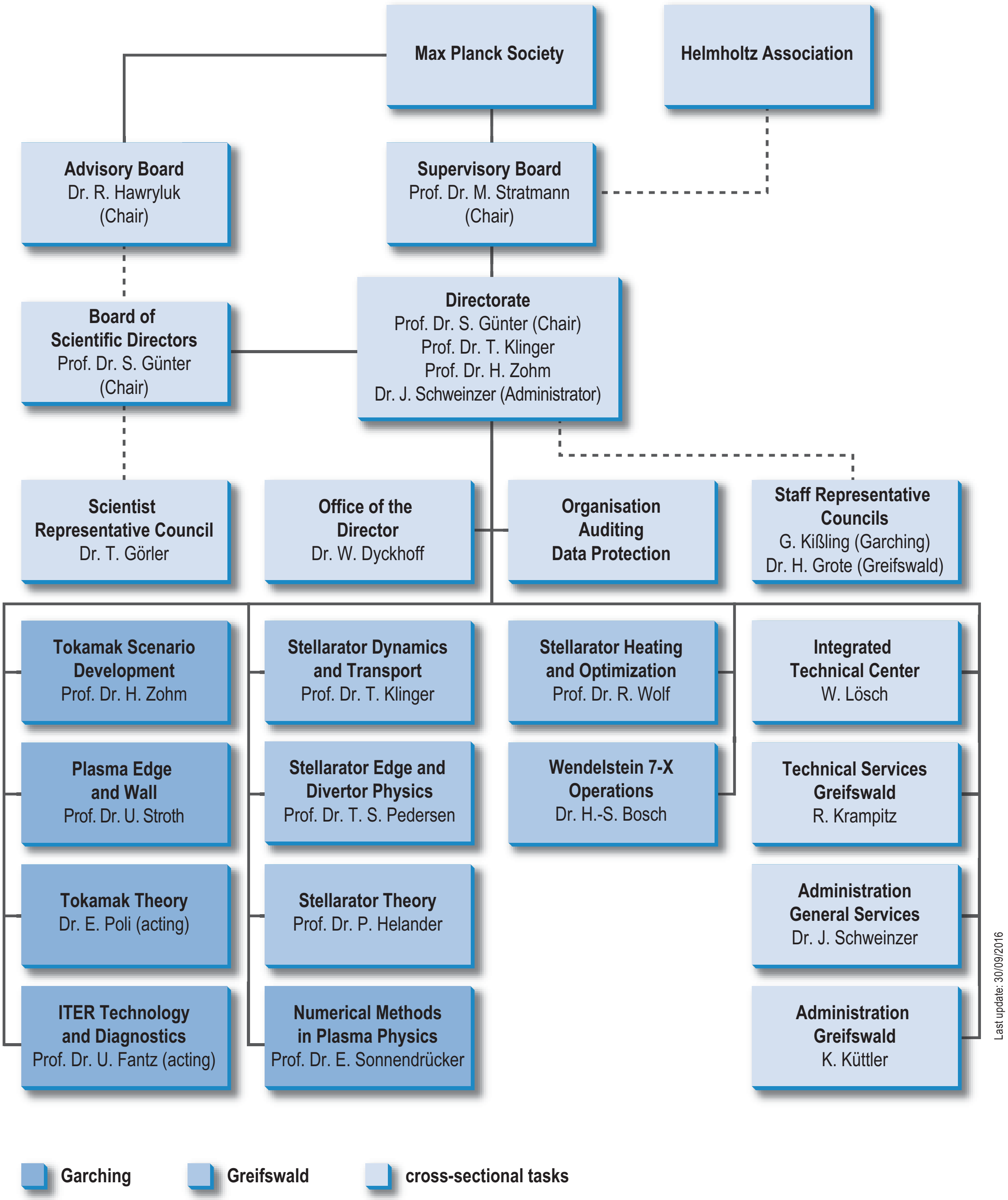
Funding

In 2015 IPP received approx. 10% of its total funding from the European Union. Of the basic national funding 90% is met by the Federal Government and 10% by the states of Bavaria and Mecklenburg-West Pomerania. European and national funding amounted to approx. 123 million euros.

Scientific Staff

At the end of September 2016 IPP had a total of 1.124 members of staff, 420 of them worked at IPP's Greifswald site. The workforce comprised 245 researchers and scientists, 82 postgraduates and 97 postdocs.

Organisational Structure



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