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Book of abstracts

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Book of abstracts

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Conference Program

Recent Progress of Japanese heliotron, LHD and Heliotron J

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The “heliotron” concept which utilizes external coils to generate magnetic field to maintain the equilibrium has been developed in Japan. The Large Helical Device (LHD) in National Institute for Fusion Science is a typical helical heliotron which succeeds the traditional concepts invented in Kyoto University about a half century ago. On the other hand, the Heliotron J device in Kyoto Univ. is an advanced helical-axis heliotron characterized by low magnetic shear and magnetic well. In this conference, recent results from both devices are presented, together with their future plans.

In LHD, finalization of the hydrogen experiments towards the deuterium experiment is going on. In order to see the effect of deuterium, it is crucially important to know the potential of the hydrogen plasma. Some trials to extend the parameter regime with hydrogen plasma have been performed in the last experimental campaign. As for the plasma heating, two MW-class 154 GHz gyrotrons have recently been installed in LHD. In addition, fine tuning of antennas to inject microwaves was also performed to optimize the power deposition on the resonant surface, using a newly developed ray-tracing code. The upgrade of the ECH scheme resulted in the achievement of the central electron temperature of 10 keV with the averaged electron density of 2×10^{19} m^{-3}. Simultaneous achievement of high ion and electron temperatures was also achieved with the formation of the electron internal transport barrier (e-ITB). Ion ITB, however, was not observed. In order to explain the result, transport analyses are being performed. Effect of ion species on high-T_i discharges were also investigated, changing the helium ion concentration in the hydrogen plasma, and preliminary results will be presented at the conference.

In Heliotron J, an improved Thomson scattering system has revealed the precise structure of the e-ITB during the ECH discharges. It is found that the threshold electron density for the formation of the e-ITB is different from that previously observed in other devices. The different threshold density is discussed in connection with the different magnetic field structure, i.e., three-dimensional effect, which is one of the most interesting topics for stellarator/heliotron systems. For the extension of the operational regime of Heliotron J, a supersonic molecular beam injection (SMBI) has greatly contributed to the increase in density range. It is considered that the SMBI can fuel in the core region without increasing the edge neutral density. Such a fuelling control technique to obtain the high performance plasma will be also tried with a newly installed pellet injector. As for the MHD studies, sophisticated fluctuation diagnostics, e.g., beam emission spectroscopy, reveals the fine structure of Alfven Eigen modes. Energetic particle modes are stabilized by using electron cyclotron current drive which changes the magnetic shear.

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Overview of TJ-II, TJ-K and H-1NF experiments

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TJ-II results. First direct observation of electrostatic potential variations within the same magnetic flux surface are presented and the influence of ECRH on fluctuations, radial electric fields and zonal flows for impurity control is discussed. The time dynamics of low frequency zonal flows and the leading role of plasma turbulence is emphasized by the observed temporal ordering of the limit cycle oscillations at the L-I-H transition. Comparative studies in tokamaks and stellarators have provided experimental evidence for the importance of multi-scale physics to unravel the impact of the isotope effect on transport. Novel solutions for plasma facing components based on the recently installed Li-liquid limiters (LLL) have been developed, showing the self-screening effect of evaporating liquid lithium protecting plasma-facing components against heat loads. Magnetic well scan experiments show that traditional stability criteria may miss some stabilization mechanisms. Further effects of ECRH on Alfvénic instabilities are investigated, showing that moderate off-axis ECH power deposition modifies the continuous nature of the Alfvén eigenmodes.

TJ-K results. Multi-probe arrays are used to diagnose plasma edge instabilities and study, e.g., their dependence on magnetic field topology. By means of two 64-pin arrays arranged in topologically different poloidal cross-sections, the normal and geodesic curvature drive of turbulent transport could be shown. The zonal-flow (ZF) / transport system shows a predator/prey relationship, with a decrease in transport following an increase in ZF amplitude. A dedicated 128-pin array measures the Reynolds-stress (RS) drive of ZFs. In wavenumber space, ZFs were found to tap energy non-locally from drift-wave turbulence, which corresponds to vortex thinning as observed in 2D neutral fluids. Poloidally, the RS is distributed asymmetrically in the same way as turbulent transport. In fact, maximum RS is found to coincide with maximum transport. To what extent field-line curvature constitutes a source channel as opposed to the geodesic transfer into the geodesic acoustic mode as a sink for ZF energy is an open issue.

H-1NF results. An Alfvén heating antenna is being installed, adapted from a Uragan design. New diagnostics include a 21 channel imaging density interferometer and an imaging helium line ratio temperature diagnostic. Tomography of three views of carbon ion optical emission from H-1 plasma reveals the 2D structure of density fluctuations in agreement with the mode structure of BAEs predicted by the CAS3D MHD code. The first images of propagating radiofrequency waves were obtained by an RF-synchronised camera viewing the H-1 plasma under the RF antenna. The observed waves have a wavelength smaller than the fast wave and are strongly affected by the position of the ion-cyclotron resonance layer. Langmuir probe correlation studies over a range of frequencies and fluctuation types reveal poloidally extended structures in both edge density and potential fluctuations at low frequency, resembling zonal flows. Configuration studies near the 7/5 resonance have shown that when the resonance is located near the plasma edge in a magnetic hill region, the plasma density exhibits strong dithering type behaviour as well a decrease in edge density.

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Recent results and program on the HSX device

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Modulated ECRH experiments in the HSX stellarator (R=1m, a=.12m, B=1 T, < 200 kW ECH) show stiffness in the electron heat transport comparable to other stellarators ($\chi_{\text{Heat pulse}}/\chi_{\text{Power balance}} \sim 1$) contrasting the case typical for tokamaks. HSX was designed through coil optimization to provide reduced neoclassical transport (comparable to a tokamak), and thus is dominated by turbulent transport. These stiffness results have been compared to gyrokinetic calculations in a flux-tube geometry using GENE. Linear simulations show the ETG mode may be relevant at the core, while the TEM mode dominates across most of the plasma – and non-linear calculations of the saturated heat flux driven by ETG and TEM bracket the experimental heat flux. TEM mode dominance in transport is suggested as the ETG mode should be stabilized with increasing density gradient, a result born out through fluctuation analysis using the nine-chord interferometer array which shown the fluctuation amplitude increasing with increasing density gradient, but decreasing with increasing temperature gradient. Fluctuation radial and poloidal velocity correlation measurements, $V_r V_\varphi$, using edge probes, have provided evidence that the Reynolds stress may play a role in the momentum balance over the neoclassical flow drives, although high and low field measurements predict higher flow drive than seen experimentally. New experiments have been performed to improve the confidence in plasma flow and electric field determination with differential-view Pfirsch-Schluter flow measurements. These results do not support the presence of the high core electron-root field predicted by the PENTA transport code.

A significant improvement in the equilibrium reconstruction of plasma current and pressure profiles has been made, resulting in the design of a new diagnostic b-dot array which has a subset optimized for fast time scale evolution of the Pfirsch-Schluter current profile reconstruction, while others are predominantly sensitive to the slower-evolving bootstrap current. Measurements made under HSX plasma operation showed greatly reduced uncertainty in the reconstruction over un-optimized diagnostics. The appearance of a rational iota was also monitored as the bootstrap current drove the core iota through 1 when operating in a magnetic hill configuration.

Plasma measurements of density, temperature, flows and potentials, in and around the edge 8/7 island structure, and when HSX is operated with a limiter inserted close to the natural separatrix have been made. Work is also progressing on examining the interface between the core and SOL particle balance at the last closed flux surface, through neutral gas and instrumented limiter probe measurements, coupled with visible and IR limiter views. These experimental measurements are compared to the edge modelling provided by EMC3-Eirene code and the neutral fuelling DEGAS code.

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15 years of construction of Wendelstein 7-X

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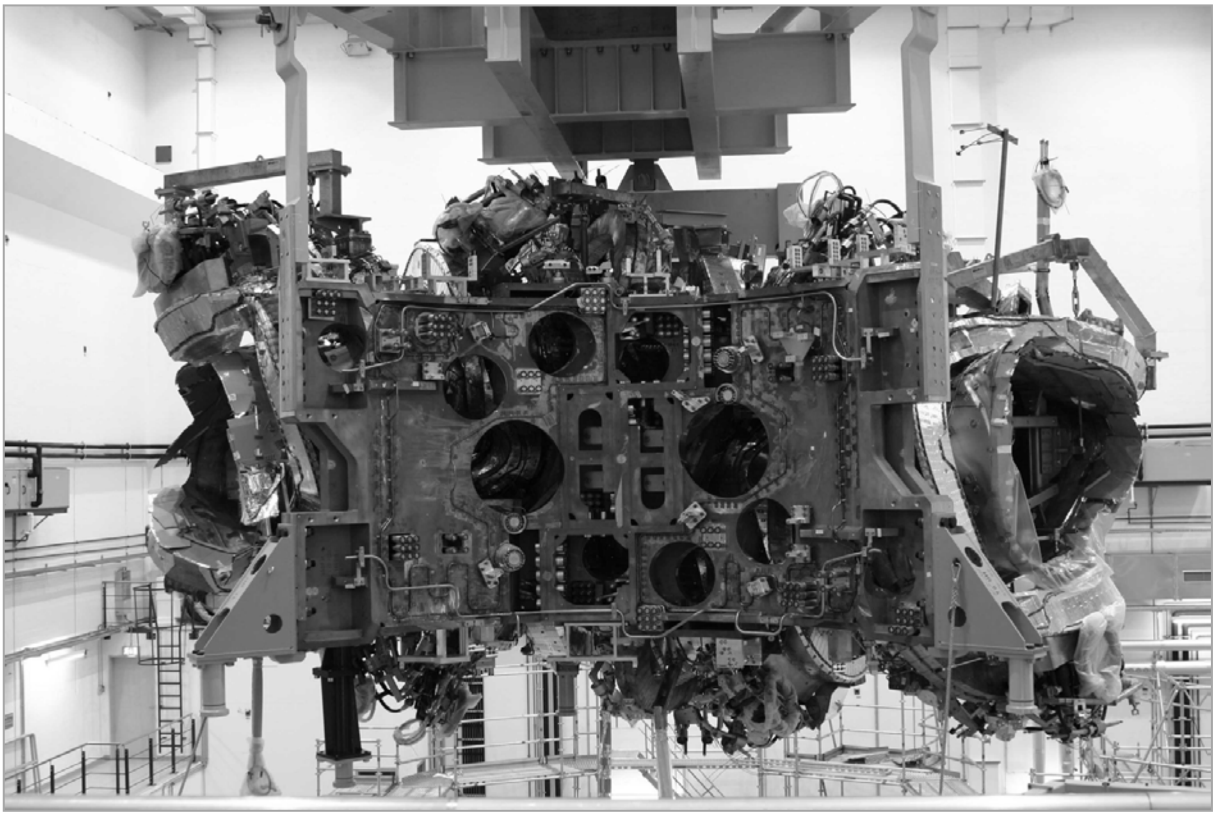
This paper summarizes the past 15 years of construction of the superconducting optimized stellarator Wendelstein 7-X. The device is now in the commissioning phase and first plasma operation is being prepared. Wendelstein 7-X is among the biggest and most advanced fusion research devices in the world and its construction was a major challenge for European industry, partner institutions, the host institute, and last no least the local construction team. We review the evolution of the construction project from the beginnings to the present with emphasis on the most important engineering issues, tools and solutions. Resulting operation guidelines and limits are discussed as well as design improvements for future devices of the Wendelstein type. The last step of the construction process is the device commissioning, a complex process mainly consisting of evacuation, cool-down and magnet ramp-up. The different commissioning steps are summarized and the current status is presented.



Completed magnet module ready for assembly as seen from the torus center. © IPP

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Session 1

Impacts of magnetic topology / 3D effects

3D Effects in Tokamaks

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Despite their nominally axisymmetry toroidal configuration, tokamaks do experience the influence of non-axisymmetric magnetic perturbations, which originate from unintentional field errors as well as consciously applied magnetic perturbations by means of suitable external magnetic field coils. Detrimental effects can occur such as the fast growth of pressure-driven MHD modes that lock to the magnetic perturbation. However, small deviations from axisymmetry with radial fields of the order of 10^{-3} - 10^{-4} times the total magnetic field can be used to the advantage of tokamak operation, for example the diagnosis of marginally stable plasma-beta driven ideal MHD core modes and the mitigation of edge-localised modes (ELMs) which are ubiquitous in high-confinement mode due to the strong pressure gradient in a narrow region near the plasma boundary. The Garching tokamak Axisymmetric Divertor Experiment (ASDEX) Upgrade is one of a few world-wide experiments equipped with a flexible set of in-vessel coils to produce non-axisymmetric perturbations. In three years of operation so far of these coils, a variety of successful experiments have been carried out, including the detection of the intrinsic error field by a compensation method, substantial ELM mitigation both at high and low plasma edge collisionality and diagnosis of the pressure-driven resistive wall mode near the ideal beta limit. In ASDEX Upgrade it is possible to control the strength of field-aligned 'resonant' perturbations independently of the presence of non-resonant spectral components, and hence resonant effects (island formation) can be separated from non-resonant effects (neoclassical toroidal viscosity, beta-driven amplification of externally induced ideal modes). In most hot high-confinement mode plasmas the resonant perturbation is very effectively shielded by electron flows, and the generation of macroscopic magnetic islands is typically not observed. ELM mitigation at low plasma collisionality depends crucially on the presence of non-resonant field components close to the plasma boundary that are amplified by the edge pressure gradient that also drives ELMs. The precise mechanism for ELM mitigation is still unknown. Stochastic edge transport that maintains the edge pressure gradient below ELM stability limits has been suspected, but can be effective only if the strong resonant field shielding is overcome, e.g. by a sufficient degree of toroidal coupling of beta-driven modes to field-aligned (resonant) modes. It is also conceivable that the ideal plasma deformation due to non-resonant modes strongly affects ELM stability. Various hypotheses will be discussed in view of the latest experimental results obtained in ASDEX Upgrade and other tokamaks.

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Effect of resonant magnetic field on MHD properties in LHD plasmas

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The RMP (Resonant Magnetic Perturbation) is applied as the effective control knob to improve the confinement performance in the torus plasmas. Though the RMP was considered to always induce the island structure in the plasmas, it is sometimes shielded. The phenomena are observed both in tokamaks and helicals including LHD. The understanding of the mechanism and the knowledge of the penetration condition are an important issue. The most remarkable result on early RMP works in LHD is the observation of the clear dependence of the RMP penetration threshold on the magnetic configuration (magnetic axis location and plasma aspect ratio) [1,2]. The property has not been reported on the tokamaks. The probable explanation is that the damping of the poloidal flow (determined by the competition between the RMP driven electro-magnetic torque and the neoclassical viscosity), which is a key parameter of the RMP shielding, has a configuration dependence. The explanation is consistent with the magnetic axis location dependence[2], but the validity for the aspect ratio has not been clarified.
Recently, the research on the threshold dependence of the m/n=1/1 RMP on the collisionality for the various configuration is progressed, and we obtain the following results; the penetration threshold increases as the collisionality in the low aspect plasmas, but the threshold decreases as the collisionality in the high aspect plasmas, and the different toroidal flow behaviour during the RMP penetration is obsebed for different aspect ratio; the change of the toroidal flow is clear in the high aspect plasmas, but it is not clear in the low aspect plasmas. The behaviour would be related with the resonant surface location. The surface of the higher aspect plasma is closer to the core region, and the toroidal flow can be larger at the surface is closer to the core in LHD. The investigation on the above RMP penetration threshold dependence would lead to the understanding of the RMP shielding and perturbation mechanism on the tokamak plasmas, where the toroidal flow is considered an important key parameter on the plasma response on the RMP. And the understanding of the RMP shielding mechanism is also the important key issue to investigate the stabilization mechanism of the interchange mode without the RMP penetration in LHD[3].

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3D edge transport modelling on tokamaks and stellarators

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Fluid plasma edge transport modelling plays an important role in interpreting data and understanding the results of existing experiments, and is a critical component of predictive simulations for future devices. As the impact of nonaxisymmetric effects in tokamaks has been widely recognized, the use of 3D models, many originally developed for stellarators, has become increasingly prominent. Applied and intrinsic 3D magnetic fields, 3D plasma facing components, and toroidally and poloidally localized heat and particle sources are all effects that must be considered. This talk will discuss application of the 3D plasma/neutral edge transport code EMC3–EIRENE to simulate 3D effects in several tokamaks. Experiments on NSTX studied how 3D fields affect plasma detachment, and showed that detached plasmas can be reattached after their application. Simulations showed qualitative agreement, with the plasma detaching at a higher upstream density with 3D fields due to short connection length flux tubes carrying hot plasma to the divertor at large major radius resulting in a striated heat flux profile. On DIII–D, the unique 2D Thomson scattering and soft X–ray diagnostics were used to test the diffusive transport model in EMC3 on different background magnetic field models. Finally, on Alcator C–Mod a series of experiments explored asymmetries induced by toroidally and poloidally localized divertor impurity gas injection. Simulations indicate that the resulting asymmetry in the divertor conditions is smaller in H–mode as compared to L–mode due to impurity ionization in the private flux region. In each case different aspects of the transport model in EMC3–EIRENE are tested, with the qualitative trends aiding understanding of the underlying physics, and the quantitative comparison pointing to areas where the model must be further developed. The role of stellarator experiments and modelling will also be discussed, in particular how measurement of the fluxes to the plasma facing components can be used to test the equilibrium magnetic field models and the balance between parallel and perpendicular transport in stochastic fields in the scrape off layer.

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3D magnetic field effect on electron internal transport barrier in Heliotron J

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The electron internal transport barrier (eITB) of helical devices plays an important role on plasma confinement [1]. This barrier is known to be formed due to the radial electric field and the electric field shear which are created by the bifurcation of radial electric field (E_r) with the electron cyclotron resonance (ECR) heating. The positive radial electric field formation is consistent with the electron-root solution of the ambipolarity condition for E_r of the neoclassical transport. In previous results in CHS, the barrier is easily formed in larger effective magnetic ripple configuration [2]. The barrier formation depends on the 3D effect of the magnetic field configuration through the neoclassical transport characteristics.

Recently, the phenomena that have similar characteristics as the eITB by the ECR heating have been observed on Heliotron J. The steep electron temperature gradient has been observed in the core region. Comparative study of the phenomena is carried out between CHS and Heliotron J. Figure 1 shows the density dependence of the temperature gradient in the core region and in the outside region of the steep temperature gradient. In both plasmas, when the line averaged density is lower than the threshold value, the temperature gradient in the core region increases. However, the threshold electron density in Heliotron J ($1.2 \times 10^{19} m^{-3}$ at P_{ECH}~330kW) is two times larger than that in CHS ($0.5 \times 10^{19} m^{-3}$ at P_{ECH}~130kW). This result shows the possibility of the threshold density increase by the difference of the 3D magnetic configuration. In this paper, the 3D magnetic field effects on the eITB in the Heliotron J plasma are discussed.

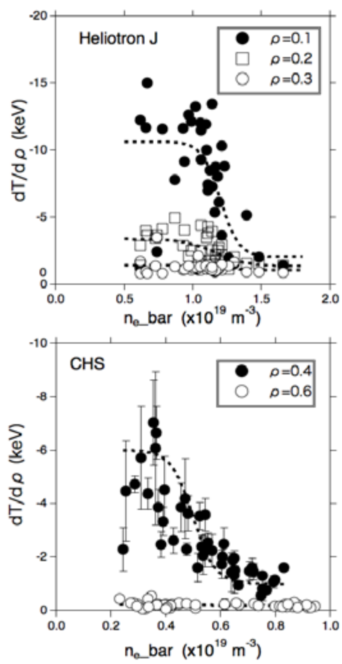

Fig.1 Density dependence on the temperature gradient in the core region and the outside of the steep temperature gradient for Heliotron J and CHS

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Validation and Verification of neoclassical transport codes for Heliotron/Stellarator devices

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Neoclassical transport (NCT) in heliotron / stellarator devices shows wide variety of nature according to the magnetic configurations. Evaluation of NC particle and energy flux together with ambipolar radial electric field is one of the important subjects to analyse the transport level in helical plasmas and to seek an optimized configuration for helical fusion reactor.
As an international collaboration in the coordinated working group meeting (CWGM), the authors have carried out a Validation and Verification (V&V) activity of many NCT simulation codes in several helical devices such as LHD, TJ-II, W7-AS etc. The main object of this research is to investigate how much difference appears among the NCT calculations according to the approximations each code adopts. The local and mono-energy codes such as DKES[1] and GSRAKE[2] neglect the grad-B and curvature drifts (magnetic drift) tangential to the flux surfaces. By comparing the local simulations with a full-5D drift-kinetic code, FORTEC-3D[3], it is revealed that the neglect of the magnetic drift term results in large error in the ion energy flux at the ion-root (negative-E_r root) in several helical devices. Since NCT in helical plasmas can become comparable to the anomalous transport, qualitatively reliable simulation is required for the study on confinement property of helical plasmas. In the presentation, we will also show how much the NC flux and ambipolar-E_r from simulations can reproduce the observation from experiments in several devices.

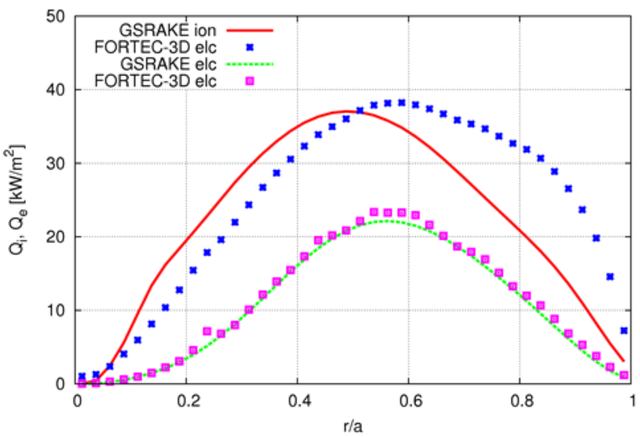

Fig: Comparisons of NC ion and electron energy flux in a LHD plasma between FORTEC-3D and GSRAKE. Large difference appears in Q_i.

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Solution of drift kinetic equation in stellarators and tokamaks with broken symmetry using the code NEO-2

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NEO-2 [1] is a linearized drift kinetic equation solver for three-dimensional toroidal magnetic fields. It has been designed in order to treat besides all other regimes effectively the long mean free path regime avoiding any simplifications on device geometry or on the Coulomb collision model. The code is based on the field line integration technique combined with a multiple domain approach, which allows introducing an adaptive grid in velocity space. This makes NEO-2 capable to resolve effectively all boundary layers between various classes of trapped particles and passing particles and also allows for a straightforward code parallelization. In stellarators, NEO-2 is used for computations of neoclassical transport coefficients in regimes with slow plasma rotation where cross-field particle motion within the magnetic surface can be ignored and, as a main application, for the evaluation of the generalized Spitzer function, which plays a role of current drive efficiency [2]. In this context, the focus is on ECCD in regimes with finite plasma collisionality. In tokamaks with small ideal non-axisymmetric magnetic field perturbations, NEO-2 is used for evaluation of the toroidal torque resulting from these perturbations (neoclassical toroidal viscosity, NTV) [3]. The limitation to slow plasma rotation pertinent to usage in stellarators has been removed in this case with help of a quasilinear approach, which is valid due to the smallness of the perturbation field. Results of these two, rather different applications as well as details of analytical and numerical problem treatment are the topic of this report. In particular, symmetry braking of the generalized Spitzer function and its effect on ECCD in W-7X is demonstrated. It appears in case of finite plasma collisionality but is absent at low and high plasma collisionalities. The results of a comparison of NTV computations between NEO-2 and other numerical and analytical approaches using as an example a standard tokamak model are discussed and such computations for ASDEX-Upgrade are presented.

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3D active modulation of edge topology in toroidal devices: effects on turbulent electromagnetic filaments

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Filament or blob structures have been observed in all magnetic configurations with very similar features despite the difference in the magnetic geometry, and are believed to play an important role in convecting particle and energy towards the wall. Despite their possibly different generation mechanism, turbulent structures and Edge Localized Mode (ELM) filaments share some common physical features. The electromagnetic (EM) effects on filament structures deserve particular interest [1], among the others for the implication they could have for ELM, related for instance to their dynamics in the transition region between closed and open field lines or to the possibility, at high beta regimes, of causing line bending which could enhance the interaction of blobs with the first wall.

A direct characterization of the effects of active modification of the edge topology on electromagnetic (EM) turbulent filament structures is presented, comparing Reversed Field Pinch (RFP) and tokamak configurations. Measurements are obtained in the RFX-mod device, which allows operating in both configurations and with different equilibria and is also equipped with an advanced system for the edge magnetic feedback control.

Different case studies of actively controlled magnetic perturbations (MP) will be analyzed focusing on the filament interaction with local magnetic islands [2,3]. High frequency fluctuations, characterizing electrostatic and magnetic filament features, and the associated transport coefficients have been observed to be strongly affected by the island proximity and topology. These observations highlight a tight correlation between the small scale EM filamentary structures and the applied MP, opening the challenging possibility of active control of filaments and their related transport by modulation of the local magnetic topology.

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Topic: S1: Impacts of magnetic topology / 3D effects

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Physics of ELM control by 3D effect in tokamaks

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Active control of edge-localized modes (ELMs) by 3D magnetic perturbation fields offers an attractive method for next-generation tokamaks, e.g. ITER. The results obtained from various tokamaks have shown that magnetic field perturbations can either completely suppress ELMs, trigger small ELMs during ELM free periods, or affect the frequency and size of the type-I ELMs in a controllable way, preserving good global energy confinement [1,2]. Although the physics mechanism is still unclear, experimental results from those different devices demonstrate that the magnetic topology plays a key role in plasma confinement, edge MHD stability, and interactions between the plasma and the first wall, particularly with the divertor.

To date, most ELM mitigation/suppression experiments have been performed on devices with a carbon-wall (C-wall). However, a carbon-free tungsten divertor is foreseen for the operation phase with activation on ITER, while the main chamber blanket modules in ITER are protected by shaped (limiter-like) beryllium panels. Recently, mitigation of type-I ELMs was observed with an n = 2 field on JET with the ITER-like wall (ILW) [3]. Several new findings with the ILW were identified and contrasted with the previous C-wall results for comparable conditions. In the moderate collisionality type-I ELMy H-mode plasmas with the ILW wall, a saturation effect of ELM mitigation and a reduction in the maximal ELM peak heat load, due to the splitting of the outer strike point, were observed during the application of n = 2 fields. Clear pre-ELM structures were observed on the outer divertor plate during the application of n=2 fields depending on q_{95}.

Recent results from EAST show that lower hybrid waves (LHWs) provide an effective means of mitigating or suppressing ELMs by inducing a profound change in the magnetic topology, similar to the effect previously observed with RMPs [4]. This has been demonstrated to be due to the formation of helical current filaments (HCFs) flowing along field lines in the scrape-off layer induced by LHWs. The change in the magnetic topology has been qualitatively modelled by considering the HCFs in a field-line-tracing code.

In this paper, a new ELM mitigation method using LHW and a comparison of the ILW results with the C-wall observations from JET will be presented. An overview of recent understanding the physics of ELM control by 3D effect in tokamak will be discussed.

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Influence of magnetic configuration on edge turbulence and transport in H1 Heliac

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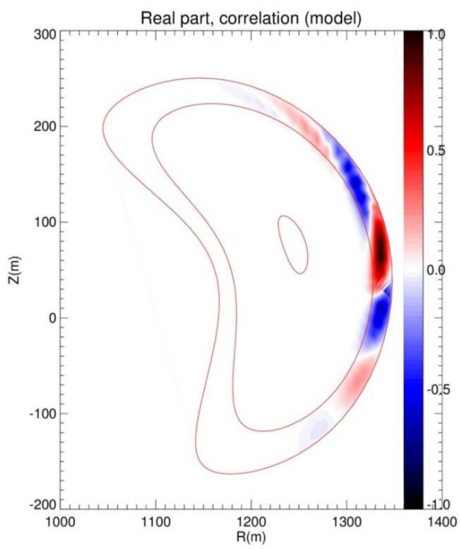
Understanding the role of magnetic islands and ergodicity on confinement and turbulence is vitally important for understanding the suppression of edge localised modes via 3D fields in Tokamaks, and Stellarators. For example, resonant magnetic perturbation coils in Tokamaks are known to produce a density pump out and modify turbulence behaviour. The fixed, but finely tuneable magnetic configuration and relatively low beta of the H-1 Heliac Stellarator device is ideal for studying in detail the role of the magnetic configuration, including rational surfaces, magnetic shear, curvature and islands on turbulence at the plasma boundary. For example, changing the current in the helical coil can produce changes in rotational transform continuously, with low shear profiles, from about 1 to 1.5. In particular, previous results show that when the edge rotational transform is close to 7/5, a pronounced drop in the density can occur (globally), and the configuration is ideal for studying the effects of edge ergodicity on confinement. Recently, scans have been done in rotational transform on a shot-by-shot basis, as well as continuously during the discharge, and drops in density can be seen in both datasets. At particular values of rotational transform, the plasma density exhibits strong dithering type behaviour with a similar temporal pattern to that of ELMs. Some shots seem to suggest that the edge density decreases strongly, characteristic of flattening in the 7/5 island or ergodic region.

In H1, various coherent modes have been previously identified as shear and beta-induced Alfven eigenmodes, however there exists a broadband part of the spectrum which is thought to be due to drift waves or may be also due to resistive interchange modes driven by the magnetic hill arising from the current being outside the plasma volume. Langmuir probes based on tungsten /Al_2O_3 ceramic in various configurations including a ball-pen probe configuration have previously been used to characterize the plasma potential, electron temperature and density around island structures in low field helicon wave heated Ar plasmas [1]. Recently, probes have been used in H/He minority RF heated discharges, where the smaller ion gyro-radius allows for interaction with smaller islands. These probes have been used to study parameter profiles, as well as turbulence and turbulent-driven flux. The total flux has also been measured based on H_α emission and S/XB (ionizations per photon) analysis, and the turbulent-driven component is seen to be of comparable to the total flux, and in particular, in some studied configurations, appears strongly dominated by certain low frequency coherent modes rather than the broad-band components characteristic of the ion gyro-scale. There is also a significant bi-coherence amongst these modes, suggestive of non-linear interaction from the driven source..

Finally, cross-correlations between probes show the radial/poloidal structure of turbulence near the edge, exhibiting a slight eddy tilt (finite k_r) and a value of $k_\theta \rho$ of ~ 0.1 at around 100kHz for broadband modes (shown in figure to the right, which is a fit to the experimental cross-coherence between a fixed and a movable probe).



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Effect of resonant magnetic perturbations on pump-out in L-mode and H-mode plasmas

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Resonant Magnetic Perturbations (RMPs) are widely employed for ELM mitigation and thus the operation of ITER is expected to take advantage from RMPs. At the same time, however, RMPs may lead to density pump-out which, in turn, affects the operation scenarios of ITER, e.g. the L-H transition. The underlying physics requires the understanding of transport in stochastic, 3D magnetic fields. In order to address particle transport effects by RMPs in view of ITER, the ITPA group on transport and confinement has conducted a comparative multi-machine study to reflect the findings and the state of understanding in particle transport with ergodized scrape-off layer fields. Although there are inherent differences between different concepts of magnetic fusion devices, it seems that they show rather common reaction to the external fields in terms of transport. At the very edge the stochastic layer is formed, which enhances parallel transport to the wall affecting the toroidal rotation and radial electric field profile.

In L-mode plasmas the influence on the edge and core plasma is typically more pronounced, when comparing to H-mode plasmas, which is attributed to much better coupling of non-axisymmetric perturbation with the plasma equilibrium. On MAST ith n=3 RMP field during L-mode discharges line averaged density is affected by the magnetic perturbation and shows a strong variation with the main, resonant component of magnetic perturbation. This is associated with flattening of Er profile at the plasma edge. On LHD, applying n=1 seeds 1/1 island near the plasma boundary, which yields significant enhancement of particle diffusion coefficient. On W7-AS confinement was strongly affected by a rotational shear and structure of stochastic boundary with island divertor.

On DIII-D ELM suppression or mitigation with n = 3 RMPs is associated with increased particle transport. Alsp changes in toroidal rotation, radial electric field, and density fluctuation level in the pedestal are observed. The reduction of core and pedestal density is explained by significant increase of long-wavelength turbulence inside the pedestal.

On ASDEX Upgrade pump-out appears at electron pedestal collisionalities below 0.5. Although there is no clear tendency to be seen, at very low collisionalities (close to ITER-like values) the effect of magnetic perturbations on density is strongest. The pump-out shows very strong correlation with pedestal toroidal rotation, which suggest strong coupling of the external fields with plasma. Also flattening in radial electric field is observed near the separatrix for the cases, where non-resonant perturbation is applied.

As for the L-H transition, the pump-out in L-mode appears to be affected by resonant interaction of stochastic fields with large scale flows. In L-mode & H-mode plasmas, both edge and core transport is affected by RMPs, what is frequently associated with a change in the ExB shear. This work is aimed to summarize recent experimental results from ASDEX-Upgrade, DIII-D, MAST, NSTX, LHD, TEXTOR and W7-AS with stochastic boundary and its influence on pump-out.

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Performance upgrade in LHD with optimum ECRH injection using ray-tracing calculations with 3D equilibrium mapping

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We have successfully increased the electron temperature of high-ion-temperature LHD plasmas (T_{e0} ~ 8 keV and T_{i0} ~ 6 keV for n_{e0} ~ 1.3 x 10^{19} m^{-3}) not only by a newly installed 154 GHz gyrotron but also by optimization of the ECRH injection. The optimization was carried out by using the ray-tracing code "*LHDGauss*", which was upgraded to include 3D equilibrium mapping data obtained in real time during experiments. For ray-tracing calculations, the *LHDGauss* can automatically read necessary data registered in the LHD database after a discharge, such as injection parameters (e.g., Gaussian beam parameters, target positions, polarization, and time evolution of ECH power) and Thomson scattering diagnostic data along with the 3D equilibrium mapping. The equilibrium map with the n_e and T_e profiles is then extrapolated into the outside of the last closed flux surface (r_{eff}/a_{99} > 1). Mode purity, which is the ratio between O-mode and X-mode, is determined by calculating the one-dimensional full-wave equation along the injection propagating direction from the antenna to the absorption target point, where the effects of the n_e profile and magnetic shear at the peripheral region under a given polarization state are taken into account by using the extrapolated virtual magnetic surfaces. Power deposition profiles calculated at each shot and each time slice of the YAG-Thomson scattering measurement during a pulse are registered in the LHD database. Figure 1 shows that the refracted EC beam under the experimental n_e and T_e profiles and the 3D equilibrium mapping results in on-axis heating. Feedback of the injection condition for the required deposition profile on a shot-by-shot basis resulted in an efficient experimental procedure.

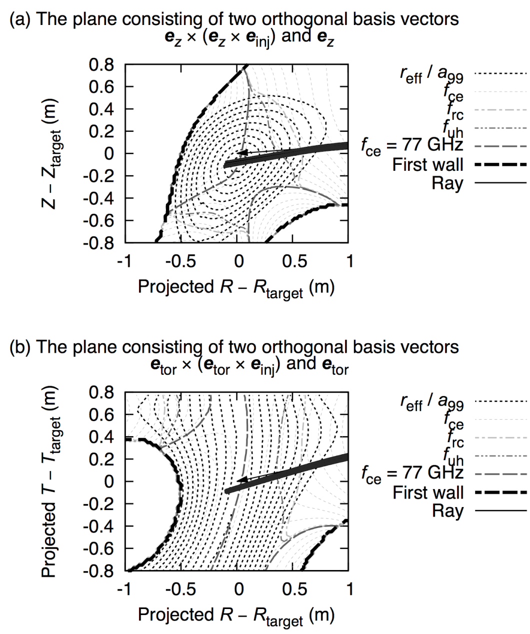

Fig. 1. Projected ray trajectories in the two planes for the EC waves injected from an outer-port launcher. Magnetic surfaces and EC resonance lines are also shown.

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Tokamak Plasma High Field Side Response to an n = 3 Magnetic Perturbation: A Comparison of 3D Equilibrium Solutions from Seven Different Codes

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In comparing equilibrium solutions for a DIII-D shot that is amenable to analysis by both stellarator and tokamak 3D equilibrium codes, a significant disagreement has been seen between solutions of the VMEC stellarator equilibrium code and solutions of tokamak perturbative 3D equilibrium codes. The source of that disagreement has been investigated, and that investigation has led to new insights into the domain of validity of the different equilibrium calculations, and to a finding that the manner in which localized screening currents at low order rational surfaces are handled can affect global properties of the equilibrium solution. The perturbative treatment has been found to break down at surprisingly small perturbation amplitudes due to overlap of the calculated perturbed flux surfaces, and that treatment is not valid in the pedestal region of the DIII-D shot studied. The perturbative treatment is valid, however, further into the interior of the plasma, and flux surface overlap does not account for the disagreement investigated here. Calculated equilibrium solutions for simple model cases and comparison of the 3D equilibrium solutions with those of other codes indicate that the disagreement arises from a difference in handling of localized currents at low order rational surfaces, with such currents being absent in VMEC and present in the perturbative codes. The significant differences in the global equilibrium solutions associated with the presence or absence of very localized screening currents at rational surfaces suggests that it may be possible to extract information about localized currents from appropriate measurements of global equilibrium plasma properties.

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Non-resonant divertors for stellarators

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Modern tokamaks have divertors that are characterized by a circular X-line along which the poloidal magnetic field vanishes. In non-axisymmetry, two distinct generalizations are possible: resonant and non-resonant divertors. For a resonant divertor, an external magnetic field is used to produce an island about a magnetic surface with a given rotational transform; $\iota = 0$ in tokamaks and $\iota = 5/5$ in W7-X. Non-resonant divertors form a sharp edge on the plasma surface, which in an axisymmetric tokamak is the X-line. A magnetic field line cannot cross a sharp edge on a magnetic surface. Sharp edges in stellarator design are helical line segments on the plasma surface. Unlike the sharp edge in a tokamak, sharp edges in a stellarator do not enforce a rotational transform on that surface. Once the line comes to the end of the edge it can encircle the magnetic surface at rate that is not determined by the edge. Magnetic field line behavior in the vicinity of a sharp edge resembles that near an X-line. A non-resonant divertor can have a stable location for the slot through which field lines to pass into a divertor chamber—independent of the plasma state. This is especially important for quasi-symmetric stellarators, which have an appreciable bootstrap current. A non-resonant divertor can also have a significant spatial separation between the region of good plasma confinement and the footpoints of the divertor field lines on the surrounding structures. This separation makes it more difficult for sputtered particles to enter the plasma volume and gives room for the beneficial loss of power by radiation. Resonant divertors tend to have the footpoint locations close to the region of good plasma confinement but have the advantage of requiring less volume within the plasma chamber.

The standard method of studying stellarator divertors is to find three-dimensional plasma equilibria with various external magnetic fields. This method is computationally intensive, which makes it difficult to understand the fundamental possibilities. A different approach will be discussed. The Garabedian representation for an outer plasma surface is known for many stellarator designs. This surface can be used to define a shape function $\vec{x}(\psi_t, \theta, \varphi)$ and a magnetic field line Hamiltonian $\psi_p(\psi_t, \theta, \varphi)$, which can be extrapolated inward and outward from the Garabedian surface is a quasi-curl-free manner. Hamilton's equations are augmented to $d\psi_t/d\varphi = D\psi_t - \partial\psi_p/\partial\theta$ and $d\theta/d\varphi = \partial\psi_p/\partial\psi_t$, which causes the field lines to spiral outwards when $D > 0$. The most important footpoints of magnetic field lines on the chamber walls are the lines just outside the last good magnetic surface, and they are obtained in the limit $D \to 0$. The spread of the loads on the divertor can be exponentially sensitive to diffusion when the magnetic field lines are stochastic. This can be investigated by letting $D = \pm D_d$, where $\pm$ is a random sign.

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Influence of resonant magnetic perturbations on transient heat load deposition

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One important aspect in tokamak research is the control of instabilities that occur in the edge of the plasma and lead to a transient energy ejection that may not be tolerable for the next generation of fusion devices. Non-axisymmetric magnetic perturbation fields which are resonant in the plasma edge are found to modify the plasma and reduce the impact of the so called edge localized modes (ELMs). Therefore, it is crucial to understand in detail how resonant magnetic perturbation (RMP) fields affect the heat load deposition.

Throughout this work, the use of fast divertor infra-red thermography at JET with the ITER-like wall shows that the transient heat load depositions on the outer horizontal target change from a random pattern to a static split structure if RMP fields are applied during a type-I ELMy H-mode. Comparisons with a thermoelectric current model show that the splitting found experimentally during ELM crashs can be explained by considering additional currents in the edge of the plasma.

Besides a modified heat load distribution, a temporally varying process is observed that appears to be linked to the static heat load structure and causes a slow propagation of a heat flux pattern long before the major energy is ejected. This process is seen for the entire q_{95} range (3.1 – 4.5) studied and slow compared to typical ELM time-scales. It appears that due to the RMPs, the process leading to a temporally varying heat flux pattern decouples from the major energy deposition.

Furthermore, this contribution discusses an approach for applying RMPs using electro-magnetic waves. Observations on the Experimental Advanced Superconducting Tokamak show strong evidence that lower hybrid waves (LHW) can create RMPs. This is seen through splitting of the outer strike-line and the successful control of ELMs while LHWs are applied. The hypothesis is strengthened by the detection of non-axisymmetric currents. For a further understanding of the features of such wave-induced RMPs, a theoretical study by implementing helical current filaments in the plasma edge has been performed. The developed model improves the theoretical understanding of the physics of LHW-induced magnetic perturbations.

* See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia
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3D magnetic geometric effects during RMP application and comparison to measurements in DIII-D *

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The physics mechanism behind ELM suppression with the application of 3D resonant magnetic perturbations (RMPs) in tokamaks is not yet well understood. The same is true for the more general observation of density pumpout during RMP application. One possibility that has been proposed invokes changes to the plasma equilibrium shaping that may destabilize microinstabilities, thereby increasing transport [1]. This work explores this possibility by calculating 3D equilibria using VMEC on typical RMP discharges on DIII-D, both with and without ELMs. Additionally, the reconstruction tool V3FIT [2] is used to find the most likely VMEC equilibrium based on a suite of diagnostic measurements. These equilibria are then used to calculate the local geometric quantities of the magnetic field that are relevant for microinstabilities, such as the curvature and local shear. A synthetic diagnostic has been developed to compare the equilibrium changes with 2D Soft X-Ray (SXR) emission measurements. Using these measurements in a phase-differenced technique whereby the RMPs are modulated by 60 degrees in the toroidal direction, the 2D SXR emission shows clear helical structure. Broadband density fluctuations measured by beam emission spectroscopy also show changes in magnitude with RMP phase, in support of the theory that microstability changes with the magnetic geometry. Finally, a scan of 3D equilibria over a large range of DIII-D parameter space has been preformed, varying these magnetic geometry quantities as much as possible in order to map out an operating space of the microstabilty mechanism. This work will ultimately allow direct comparisons with 3D gyrokinetic equation solvers to validate the microstability mechanism, but these codes will first need to be upgraded to enable use in non-stellarator-symmetric, diverted H-mode tokamak geometries, which is the typical operating regime for ELM suppressed RMP cases.

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Effect of 3D magnetic perturbations on the plasma rotation in tokamaks

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Plasma rotation in tokamaks can be significantly influenced by the torque arising from radial plasma currents due to 3D magnetic perturbations (e.g., toroidal field ripple, error fields, coils for ELM mitigation purposes) [1]. Since this torque is often expressed through a viscous force, the phenomenon is termed as neoclassical toroidal plasma viscosity (NTV). For the evaluation of the NTV torque several analytical and semi-analytical approaches [2] are presently used, which make simplifying assumptions concerning geometry and collision operators. A numerical approach without such simplifications is provided by the upgraded version of the code NEO-2 [3]. The only assumption is that the perturbations are small enough such that the particle motion within the perturbed flux surface is only weakly affected by the perturbation field (quasilinear approach).

In the present report the NTV torque is evaluated numerically using the upgraded code NEO-2 for a realistic tokamak equilibrium (ASDEX Upgrade) and a single-species plasma. It is found that the torque from the TF ripple and the torque from ELM mitigation coils are localized at the outer plasma region and are of the same order (about 1 Nm). The torque due to the TF ripple (short scale perturbation, n=16) is mainly applied to ions and is well described by the analytical formula for the ripple-plateau regime [4], whereas the torque from the ELM mitigation coils (medium scale perturbations, n=2,6) contains significant contributions from both, bounce-averaged regimes and from the resonant regimes. The collision model (full linearized collision operator), which has been implemented so far in NEO-2 for a simple plasma (electrons and one sort of ions), has been generalized here for the general case of a multi-species plasma. This would allow computations of the torque for plasmas with significant impurity contents including the computations of the additional impurity tranport caused in tokamaks by the violation of the toroidal magnetic field symmetry.

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Toroidal torque in resonant transport regimes of tokamak plasmas with perturbed axisymmetry within Hamiltonian theory

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We are investigating the effect on toroidal plasma rotation by non-resonant non-axisymmetric magnetic perturbations, such as toroidal field ripples, error fields and external magnetic perturbations for ELM control in axisymmetric plasmas. This rotation is changed by non-ambipolar neoclassical transport caused by the perturbations, an effect known as neoclassical toroidal viscosity (NTV). In the low-collisionality limit, collisionless transport is generated by resonant particles.

When the collision time exceeds orbital timescales, resonances between bounce frequency of trapped particles (or the transit frequency of passing particles) and the cross-field drift due to the perturbation become possible (drift orbit resonances). If $\mathbf{E} \times \mathbf{B}$ and ∇B drift balance each other, the resulting resonant regime is the superbanana plateau regime. The small fraction of particles that fulfill resonance conditions is expected to dominate neoclassical radial particle transport in comparison to non-resonant particles in low-collisionality regimes [1-3].

The Hamiltonian action-angle formalism offers a way of describing resonant regimes in a unified way. We are addressing both superbanana plateau and drift orbit resonances in the quasilinear approximation [4]. Derivations of expressions for toroidal torque and radial particle flux and a numerical implementation are presented. Results from this approach are compared to existing analytical results and numerical calculations with the NEO-2 code [5,6].

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Effect of the RMP on the ELM-like activities observed in the high-beta H-mode of LHD

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Formation of the edge transport barrier is the key to achieve high confinement plasma in Tokamak plasmas. H-mode transition in the Large Helical Device (LHD) was also realized in relatively low beta condition, beta up to 2.5%, with the vacuum magnetic location Rax = 3.6m~3.9m [1]. In the last experimental campaign, operational regime in more inward shifted configurations, e.g., Rax = 3.56m, is investigated [2], H-mode transition with higher beta = 3.0% (1T) ~3.5% (0.75T) was clearly observed in this regime and this transition is necessary to achieve plasmas with good confinement. With the transition, the rapid increase in the electron density is observed in the very peripheral region. That region is definitely outside the plasma boundary or the last closed flux surface of the vacuum magnetic field. After the transition, strong ELM-like edge MHD instabilities localized in the magnetic stochastic layer are activated from the increase of the pressure gradient, as shown in Fig. 1. The increase of the stored energy is limited by this excitation of the edge MHD instabilities. Coherence between the magnetic fluctuations and the fluctuating component of the ion saturation currents measured at the divertor plate also suggests that the ELM-like activities are responsible for enhancing the transport in the peripheral region. Application of the m/n = 1/1 RMP field is performed for controlling the ELM-like activities. Though the effect on the amplitude of the MHD activities is not large, a reduction of the fluctuating component of the particle flux onto the divertor is observed. When the RMP field is large so that m/n = 1/1 island is formed, the H-mode cannot be achieved.

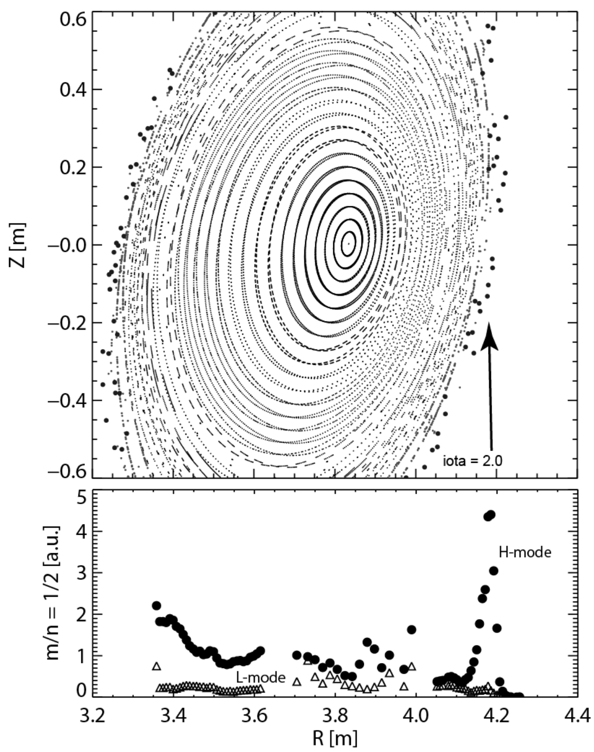

Fig. 1: Poincarre map of the magnetic field estimated by HINT2 and the fluctuation amplitude of the m/n = 1/2 MHD instabiliities measured by the CO_2 laser imaging interferometer.

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3D topology of RFX-mod edge

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The edge of magnetic confined plasmas is often characterized by a 3D topology due the distortion of magnetic flux surfaces along the toroidal and poloidal directions shaped by the presence of magnetic islands. These islands are created purposely in the tokamak to stabilize edge localized modes (ELMs), while they are spontaneously present in reversed-field pinches (RFPs) and stellarators.

In the RFX-mod (R=2m, a=0.46m) RFP, high current plasma regimes (I_ϕ>1MA, n/n_G<0.35, with n_G the Greenwald density) are characterized by the resonance of an m/n = 1/7 tearing mode in the plasma core (r/a~0.3), that molds all plasma column in a 3D helical topology [1]. The resulting edge magnetic ripple (~1%) is enough to modulate edge floating potential, electron density and temperature, particle flux and flow. The presence of this magnetic island in the central region produces, thanks to the coupling with the Shafranov shift, a chain of "remnant" islands (m/n = 0/7) on the q = 0 surface in the edge region, where the typical m = 0 resonances develop. The resulting magnetic topology is quite complex: in the low-field side (LFS) the m = 0 O-point (OP) is aligned toroidally with the m = 1 X-point (XP), the converse in the high-field side (HFS), where the m = 0 OP is aligned with the m = 1 OP [2]. A useful tool for this analysis is the introduction of a frame of reference travelling with the dominant mode, i.e. the helical angle $u = m\theta - n\varphi + \omega t$ [1], in order to correctly relate different toroidal and poloidal measurements with respect to the plasma helix. Considering measurements resolved both in the toroidal and the poloidal directions (floating potential and electron density and temperature), the results indicate that the modulation in the edge plasma is not a simple 1/7 harmonic but the shape of plasma response shows the presence of sidebands in the (m,n) spectrum.

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3D Equilibrium Reconstruction for H-mode Tokamaks

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The application of resonant magnetic perturbations for edge localized mode (ELM) mitigation breaks the toroidal symmetry of tokamaks. As a consequence the axisymmetric assumptions of the Grad-Shafranov equation and the tools build around those assumptions cannot capture the full physics of the perturbed equilibrium. The VMEC[1] equilibrium code makes no such assumptions and can solve the equilibrium state of a wide variety of 3D systems (Figure). In these systems the 3D structures of the equilibrium are very pronounced. By contrast the 3D perturbations in tokamaks are on the order of a 1% or less.

This presents a challenge for 3D equilibrium reconstruction of tokamaks. When minimizing the mismatch between the experimentally observed signals and the modeled signals, the weakly 3D components can get lost in the noise. In this presentation will cover the adaptation of the 3D equilibrium reconstruct code V3FIT[2] for the application to 3D perturbed DIII-D tokamak. Among these enhancements to V3FIT are:

- The ability isolate 3D components of signals from the axisymmetric solution.
- Motional stark effect signals for reconstruction current and q profiles.
- Parallelization of VMEC and V3FIT to make use of super computing resources.

In addition, a work flow starting from a 2D reconstructed equilibria to a fully 3D reconstruction will be discussed.

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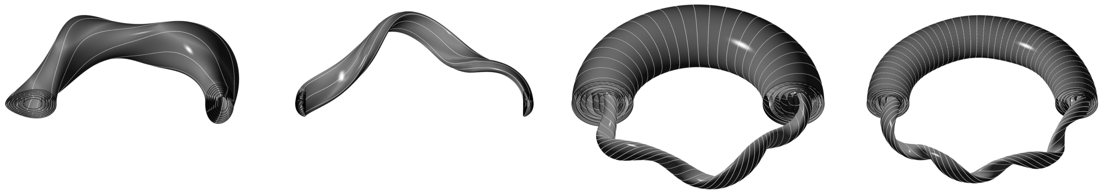

Reconstructed equilibria for, from left to right, CTH, HSX, MST and RFX-Mod.

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The tokamak: a frustrated stellarator

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The tokamak system is designed to retain axisymmetric properties to a very high degree which is an important ingredient for enhanced neoclassical transport properties and improved energetic particle confinement. However, when plasma current and $\langle\beta\rangle$ limits are approached and toroidal current and pressure profiles are tailored either to avoid a low order mode rational surface ($\iota \geq 1$ and sawteeth) or improve confinement (H-mode), saturated kink structures can develop that insinuate that the tokamak prefers to evolve towards a stellarator state. Long-lived Modes (LLM) in tokamaks like MAST [1] under hybrid scenario conditions with weakly reversed central magnetic shear ($\iota_{max} \simeq 1$) and Edge Harmonic Oscillations (EHO) [2] observed in Quiescent H-mode discharges in DIII-D are associated with 3D structures that we can now recover as nonlinearly saturated ideal internal and external kink modes, respectively, with the 3D VMEC equilibrium code.

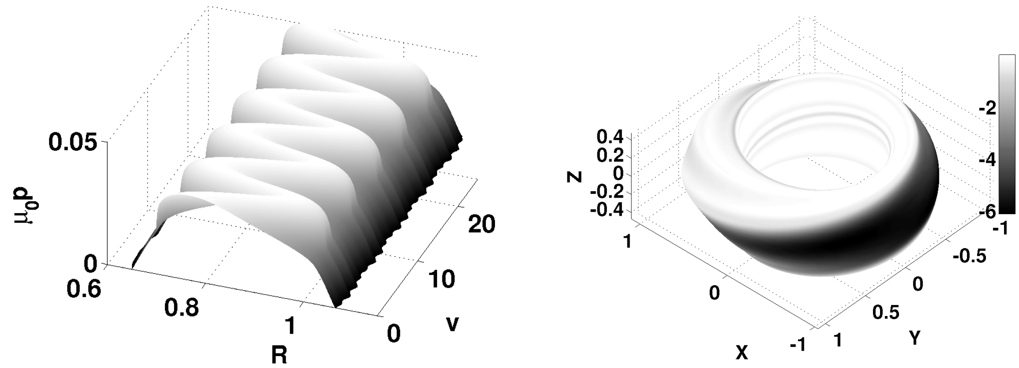

Fig. 1. (Left) The pressure at the midplane over 5 toroidal transits constitutes a manifestation of a saturated ideal internal $n = 1$ kink. (Right) The $\sqrt{g}j_{||}^2$ distribution on a surface close to the edge of the plasma corresponds to a saturated ideal external $n = 1$ peeling/kink structure with the characteristics of an EHO or an Outer Mode (OM) [3].

Typical saturated mode structures as 3D equilibrium states computed for the TCV tokamak are displayed in Figure 1. It is quite clear from the pressure distribution $\mu_0 p$ and the parallel current $j_{||}$ shown in the figure that the plasma state is strongly 3D, hence more reminiscent of a stellarator configuration than that of a standard tokamak.

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Influence of the Trim Coils Operation on Wendelstein 7-X Magnetic Field

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Wendelstein 7-X (W7-X), currently under commissioning at the Max-Planck-Institut für Plasmaphysik in Greifswald, Germany, is a modular advanced stellarator with a magnetic field optimized for good plasma confinement and stability. Several of the planned W7-X magnetic configurations are rather sensitive to the symmetry breaking perturbations resulting from unavoidable construction displacements and manufacturing tolerances. The most critical consequences of such magnetic field perturbations are modifications of the magnetic island topology, which can lead to uneven loads on the divertor targets and affect the plasma performance.

To evaluate possible changes in the magnetic field properties, step-by-step tracking of the magnet system geometry [1] was performed accompanying the entire process of the magnet system construction. On the basis of these measurements positions of the five machine modules on the machine base were successively optimised which helped to avoid an error field accumulation during the assembly and to reduce magnetic field perturbations significantly [2]. To eliminate the residual error fields five trim coils are installed in W7-X. The electromagnetic forces due to the operation of the trim coils can in turn cause deformations of the magnet system. Possible worst case scenarios of the asymmetric magnet system deformations coming from the unbalanced force under the trim coil operation were modelled in the high iota operating regime with help of the 360° ANSYS Global Model and 360° ABAQUS Finite Element Model [3]. It was found that the trim coil operation could cause asymmetric winding pack displacements up to 4 mm. Resulting from the finite element analysis displacements of all 70 winding packs of the main magnet system were the input for the electromagnetic analysis. This paper presents the results of the electromagnetic studies and shows that in view of the capability of the trim coils to generate B_{11}/B_0 up to $6.67\cdot10^{-4}$ and B_{22}/B_0 up to $2.67\cdot10^{-4}$ for B_0 = 3T, the considered perturbations are small and can be corrected by a slight readjustment of the trim coil currents.

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Error field detection and correction on W7-X

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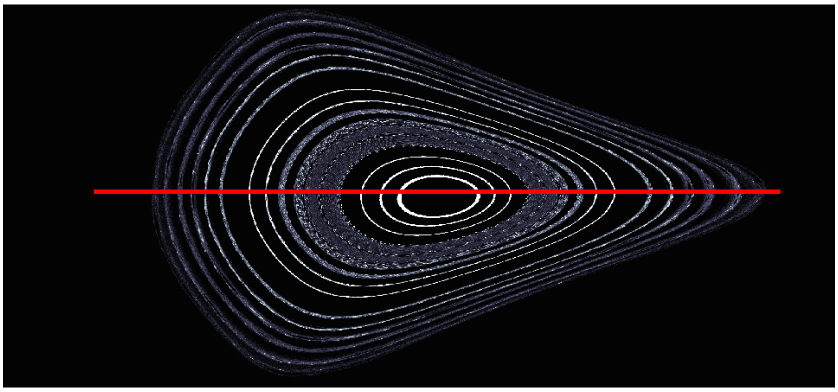
Among the first early results from the W7-X experimental program will be the measurement of flux surfaces during the flux surface mapping campaign [1]. This experimental technique will leverage a unique feature of stellarators, namely vacuum magnetic flux surfaces, to confirm the existence of flux surfaces, assess the presence of error fields, and verify the effects of the five trim coils on the experiment [2]. This diagnostic technique requires that an electron gun be placed into the vacuum vessel with the field coils energised. The path of the electrons along the magnetic field lines can then be imaged using a swept fluorescent rod [3]. In order to amplify the effect of error fields on the magnetic configuration, a configuration with rotational transform very near unity at the magnetic axis has been developed. This results in a helical excursion of the magnetic axis. The location of the axis then becomes a function of the phase and amplitude of both intrinsic and applied $n = 1$ fields. It is shown that by rotating the applied error field the intrinsic error field phase and amplitude may be determined using a fitting method. Theoretical predictions of electron beam mapping [4] are presented alongside experimental measurements. The 'as-built' coil set error field will also be compared to the experimentally measured error field.



Synthetic camera image showing vertical displacement of flux surfaces due to applied error fields (horizontal line at Z=0).

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Free-boundary Ideal MHD Stability of W7-X Divertor Equilibria

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The Wendelstein 7-X stellarator at Greifswald is currently in its commissioning phase and will be equipped with an island divertor after an initial phase operating with plasma limiters.

The positions of the edge islands, at rotational transforms of either $\iota = 5/4$ or 5/5 or 5/6, are crucial for the performance of the divertor, which limits the number of feasible configurations within the variety allowed for by the W7-X magnet system. A further restriction arises from the plasma bootstrap current, as the edge islands stay in their vacuum locations only if either it can be balanced or if it is sufficiently small. The latter case poses a constraint on the vacuum-field mirror[1].

As continuation and extension of earlier work [2], W7-X plasma configurations satisfying the divertor requirements are studied for their ideal MHD stability properties against free-boundary perturbations. The spatial structures as well as growth rates of unstable perturbations will be discussed. Special focus is laid on MHD unstable regions of the configuration space, since this may reveal scenarios in which the importance of linear MHD in low-shear stellarators can be studied experimentally.

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Modeling of limiter heat loads and impurity transport in Wendelstein 7-X startup plasmas

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The quasi-isodynamic stellarator Wendelstein 7-X starts plasma operation in 2015. During the first operation phase the plasmas will be in a limiter configuration [1],[2]. In this field configuration the plasma boundary does not include magnetic islands and the scrape-off layer (SOL) is defined by five poloidal graphite limiters located at the bean shaped symmetry planes. The limiters define the position of the last closed flux surface and are positioned such that they can prevent high heat fluxes onto the unprotected main chamber wall and metallic frame structure of the later installed graphite divertor targets. They are optimized for an evenly distributed heat load deposition. Considering startup plasmas with heating power up to 5MW and densities up to $2{\times}10^{19}\mathrm{m}^{-3}$ heat loads to the limiters and the generation of impurities due to plasma surface interaction (PSI) become a concern. In this contribution, the edge transport and PSI characteristics of the W7-X limiter startup plasmas are studied using the 3D fluid plasma edge and kinetic neutral transport Monte Carlo code EMC3-Eirene.

A connection length analysis shows that the limiters cause a 3D SOL topology consisting of three seperate helical magnetic flux bundles of different field line connection lengths. This complex topology results in separate heat flux channels featuring localized peaks in the limiter heat load deposition patterns.

A density scan at high input power of 5MW was performed to analyse limiter heat loads, impurity production and transport including radiative edge cooling. The results show a strong topological impact on the pressure and flow profiles. Increasing density yields a graduate reduction of the limiter peak heat fluxes and an increase of the heat flux channel widths λ_q. The feasibility of radiation cooling by intrinsic sputtered carbon and active injected nitrogen as a means for heat flux mitigation on the limiters is stated as a preparation of this method for the island divertor phase.

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Development of a free-boundary version of SIESTA: application to the Wendelstein 7-X stellarator

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In order to better understand the dynamics of fusion plasmas it is of utmost importance to have a reliable knowledge of the magnetohydrodynamics of the given device. SIESTA [1] is a code that allows the self-consistent calculation of the MHD equilibrium of any three-dimensional machine. Although the code is still in development, it is already being successfully applied to a number of machines. There exist other codes (*e.g.* PIES and HINT2) which do 3-D MHD analysis, but using different constraints that generally require much longer to solve for a given equilibrium. SIESTA relies on the solution found by VMEC—which imposes the existence of nested flux surfaces— for the same configuration, using it to provide a fixed background coordinate system and an initial solution. Then, it iterates the solution without imposing any restriction on the field (except at the boundary), which allows for the development of stochastic regions and magnetic islands. Up to now, SIESTA has worked as a fixed boundary code, which prevents its application to any problem requiring knowledge of the magnetic field between the plasma and the vacuum vessel. In this work, we present the first results of ongoing efforts to turn SIESTA into a full free-boundary code.

Tests are currently being done on the W7-X standard configuration. The approach followed consists of expanding the VMEC solution all the way to the vacuum vessel, which becomes the new fixed boundary. In this way, the fixed-boundary SIESTA can still be used to iterate towards equilibrium while allowing for the distortion of the LCFS. Using SIESTA in this way requires the artificial extention of the coordinate system provided by VMEC all the way to the new fixed boundary. Although these expanded coordinates are no longer flux coordinates in the region between the plasma and the vessel, this is of no importance since SIESTA uses them only as a fixed background coordinate system. In order to provide with a good initial guess for the magnetic field over the outer region, the IPP EXTENDER code [2] has been used. EXTENDER is based on the virtual casing principle [2, 3] to calculate the magnetic field created by the plasma currents calculated by VMEC, to which the contribution of the external coils is then added. Initial results obtained with the new approach for W7-X will be shown.

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A novel solution for the computation of three-dimensional ideal-MHD equilibria with current sheets at resonant surfaces

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Ideal MHD predicts the existence of singular current densities forming at rational surfaces in three-dimensional equilibria with nested flux surfaces, thus making non-smooth solutions ubiquitous to the 3D MHD problem. The singularities consist of a Pfirsch-Schlüter, 1/x current density that arises *around* rational surfaces as a result of finite pressure gradient, and a Dirac δ-function current that develops *at* rational surfaces as a necessary mechanism to prevent the formation of islands that would otherwise develop in a non-ideal plasma. Only recently have these currents been computed numerically [1], and we provide details of their calculation. We show that locally-infinite shear at the resonant surfaces is required in order to have well-defined solutions. This suggests extending the possible classes of ideal-MHD equilibrium solutions to include those with discontinuous rotational-transform [2].

Singularities in the current *density* are allowed in the ideal-MHD model, but the current passing through any surface must remain finite. While the integral of the δ-current density is always finite, certain surfaces may be constructed through which the Pfirsch-Schlüter current diverges logarithmically; thus pressure gradients cannot be supported by rational surfaces. Historically, the cause of pathologies in MHD equilibria with nested surfaces has been attributed to the possible pressure profiles, which cannot be smooth, as noted by Grad.

In this talk, we present a new class of three-dimensional, globally-ideal, MHD equilibria with discontinuous rotational-transform across the resonant, rational surfaces, with (i) continuously nested flux surfaces, (ii) arbitrary, continuous and smooth pressure profiles, (iii) arbitrary, three-dimensional boundaries, (iv) without unphysical plasma currents, and which are (v) analytic functions of the boundary. Examples of such three-dimensional MHD equilibria, computed with the SPEC code, are shown. A verification exercise is carried out in cylindrical geometry by comparing the numerical results to generalized solutions to Newcomb equation, showing excellent convergence.

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Expansion of Non-Symmetric Toroidal Ideal MHD Equilibria About a Magnetic Axis

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The construction of non-symmetric toroidal ideal MHD equilibria raises many questions. This work reconsiders the possibility of equilibria with nested flux surfaces, following earlier work, Phys. Plasmas, 21, 022515 (2014), where equilibria were found to all orders in an expansion in the magnitude of the helical field amplitudes. That work was in a topological torus, and not of direct extension to a true torus. Further, work in J. Math. Phys, 26, 2370 (1985) claimed that resonance destroyed the possibility of an expansion about the magnetic axis of any non-symmetric equilibrium in a toroidal domain.

This work re-examines the expansion about a magnetic axis of a non-symmetric equilibrium in powers of the distance from the axis in a toroidal domain and shows that in many cases it is possible to expand to all orders in that distance. It is possible to avoid the effects of resonant fields, which normally induce singular currents, by judicious choice of field components. When the axis is a circle and the flux surfaces are approximate circular cylindrical tori, the analysis is complete. Flux surface and magnetic field components, which have many derivatives but are not analytic, are used in the construction. When the flux surfaces are approximate elliptic cylinder tori it appears that the same type of construction works, but it becomes extremely complex to carry through. When the magnetic axis is a non-circular space curve the analysis is less secure but strongly suggests that no equilibria, or at best a very narrow set of equilibria are possible.

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Modeling of Island Divertor Plates in the Compact Toroidal Hybrid

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Edge island divertors can be used as a method of plasma particle and heat exhaust in long pulse length stellarator experiments. Computational studies of the power loading on these structures and its relationship to the long connection length scrape off layer physics are underway on the Compact Toroidal Hybrid (CTH) experiment, a five-field period torsatron with $R_0 = 0.75\,\mathrm{m}$, $a_p \sim 0.2\,\mathrm{m}$, and $\mathrm{B} \leq 0.7\,\mathrm{T}$. We report the results of connection length studies for divertor plates to be installed in CTH and initial calculations using the EMC3-EIRENE code[1]. For these studies, CTH will be operated as a pure stellarator with no ohmically generated plasma current. Plasma generation and heating will be accomplished with a 200kW, 2.4GHz gyrotron operating at 2nd harmonic that is under construction. CTH can be operated over a wide range of magnetic field geometries with an edge rotational transform from $\iota_{\mathrm{vac}}(\mathrm{a}) = 0.02 - 0.35$. A poloidal field coil is used to adjust the shear of the rotational transform profile, and hence the size of edge islands, while five error coils producing an $\mathrm{n} = 1$ perturbation giving further size and phase adjustment. For the studies conducted, a magnetic configuration with a large $\mathrm{n} = 1$, $\mathrm{m} = 3$ magnetic island at the edge is generated. Results from multiple possible divertor plate locations relative to the island structure will be presented and discussed. It has been found that the flexibility of the CTH system allows access to parameter space where the parallel transport effects can dominate the perpendicular effects or visa versa. The connection length studies are performed using the field line following capabilities of the IFT[2,3] code, while both IFT and the FIELDLINES codes were used to generate the input grids for the EMC3 code.

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Effect of magnetic field geometry on blob structure and dynamics in TJ-K

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Mesoscale, filamental density perturbations, elongated along magnetic field lines, known also as blobs [1], are ubiquitous in the scrape-off layer (SOL) of magnetic confinement fusion devices. Denser and hotter than the background SOL plasma, they transport particles and heat across magnetic field lines to the reactor wall. A detailed understanding of blob dynamics is therefore necessary when predicting particle and heat transport through the SOL and the potential heat loads on plasma-facing components.

Blob propagation is thought to be driven by magnetic field line curvature, with the normal curvature driving radial propagation and the geodesic curvature driving poloidal propagation. Due to experimental constraints, investigations of blob velocity are often limited to the radial component, however in TJ-K Langmuir probe access in the entire plasma volume allows the poloidal velocity component to also be studied. Using the conditional averaging technique on density fluctuations in a poloidal cross section of TJ-K, the radial and poloidal components of blob velocity have been calculated and compared to an analytical blob model [2], simplified to express blob velocity in terms of the magnetic field curvature vector. As previously observed, the radial blob velocity component is well predicted by the model, however in the case of the poloidal component, the background ExB flow must also be taken into account for the model to be comparable to experimental velocities.

Due to the field aligned nature of blobs, it is predicted that their three dimensional structure is influenced by magnetic field shear. A rather extreme example of this is the deformation of a structure as it passes near the X-point of a divertor (e.g. [3]). By conditional averaging of density fluctuations in two toroidally separated poloidal cross sections, the 3D structure of blobs in TJ-K has been investigated, and the influence of local magnetic field geometry on blob filament structure has been studied at different locations in a poloidal cross section.

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An examination of the chaotic magnetic field near the separatrix of magnetically confined plasmas

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Understanding the chaotic structure of the magnetic field near the plasma edge is important for computing the MHD equilibrium, understanding the MHD stability, and for understanding a variety of transport phenomena. All magnetically confined plasmas are surrounded by a separatrix, which for perfectly-axisymmetric tokamaks usually defines the plasma edge. As three-dimensional perturbations are applied, the stable and unstable branches of the separatrix split, the homoclinic tangle is formed, and flux surfaces near the separatrix are destroyed. For strongly-nonaxisymmetric stellarators and heliotrons, the flux-surface destruction near the separatrix is usually considerable and the boundary surface (i.e. the last closed magnetic surface) may be sufficiently far from the separatrix so that the homoclinic tangle, or heteroclinic tangle, does not require special attention in computational simulations.

Efficient numerical techniques for locating the boundary surface, the cantori and for constructing straight-fieldline coordinates (where possible) have been implemented [1] for the chaotic magnetic fields near the plasma edge in LHD, as computed by HINT2 [2]. These techniques will be extended to tokamak geometry, for which the proximity of the separatrix to the plasma edges demands additional care. The singular structure of the separatrix will be treated analytically, so that the computational coordinates are smooth and well-behaved. The interplay between the homoclinic tangle and the cantori will be illustrated.

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Advances in 3-D Equilibrium Reconstruction using V3FIT

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V3FIT[1] is a fully three dimensional equilibrium reconstruction code used to analyse a wide range of fusion devices. Built around the VMEC[2] 3-D equilibrium solver, V3FIT can reconstruct any configuration with closed nested flux surfaces. V3FIT is used to reconstruct equilibria on the Compact Toroidal Hybrid (CTH) and Helically Symmetric eXperiment (HSX) stellarators, and the Madison Symmetric Torus (MST) and RFX-Mod reversed-field pinches. On the DIII-D and JET tokamaks, V3FIT is used to explore the deformations that error correction coils apply to the nominally axisymmetric plasma. Initial forward modelling has begun for the Wendelstein 7-X experiment.

The code has recently undergone extensive revision to prepare for the next generation of equilibrium reconstruction problems. Enhancements include:

- Equilibrium model abstraction, for support of multiple equilibrium solvers,
- Addition of the VACUUM equilibrium model,
- Propagation of experimental errors to reconstructed results, and computation of posterior covariances,
- Support for multicolour soft x-ray emissivity cameras,
- Parallelization of the reconstruction algorithm, using both OpenMP and MPI,
- Support for a parallelized version of VMEC,
- Additional capability of the magnetic diagnostic model to include the effects of a thick conducting shell.

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Free-boundary equilibria for stellarator configurations given by their boundary geometry

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A non-quasisymmetric stellarator vacuum magnetic field witn good collisionless particle confinement [1] determined by its last closed magnetic surface is reconstructed by finding a remote surface-current density. This intermediate step before finding a modular coil system enables free-boundary finite-β equilibria to be calculated so that the physical properties of the configuration can be assessed and will be described for the configuration of [1].

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3D MHD simulations of stellarator plasmas with SPECYL and PIXIE3D codes

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Preliminary simulations of stellarator-like configurations with the nonlinear 3D MHD codes SPECYL and PIXIE3D are reported in this work. SPECYL [1] solves the zero-β visco-resistive MHD equations in cylindrical geometry, whereas PIXIE3D [2] can run in toroidal geometry as well and with finite plasma pressure. The mathematical correctness of the two codes was proven by a nonlinear cross-benchmark study [3]. Both codes have been used to study the nonlinear MHD dynamics of reversed-field pinch (RFP) and tokamak plasmas. Qualitative agreement with respect to experimental observations in the RFX-mod device, operated both in RFP and tokamak modes, has been demonstrated, in particular with the application of external magnetic perturbations [4,5,6]. More recently, helical perturbations of the magnetic boundary have been used in both codes to produce stellarator fields within an axisymmetric computational domain [6].

Here, we first focus on zero-β stellarator configurations in both cylindrical and toroidal geometry. The equilibrium properties such as the ι profile and the magnetic field topology are discussed. Magnetic islands and stochastic regions appear when the helical symmetry provided by the dominant helical perturbation is violated. As expected, this occurs in the cylindrical case when secondary magnetic perturbations are applied, and in the toroidal case due to geometric effects, even without secondary perturbations. The connection length of field lines to the wall provides a measure of edge stochasticity. Finite-β stellarator simulations are then considered with the PIXIE3D code in cylindrical geometry. The finite-β helical equilibria obtained with increasing heating sources are described and compared with the zero-β solution.

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Experimental characterization and equilibrium reconstructions of first electron cyclotron heated plasmas in the low-aspect-ratio CNT stellarator

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CNT is a university-scale device whose four circular planar coils confine stellarator plasmas with the lowest aspect ratio ever attained [1]. During its first decade of operation, CNT confined pure-electron plasmas [2] and plasmas of arbitrary degree of neutrality [3]. Recently, neutral plasmas started up and sustained by electron cyclotron resonance heating have been studied in CNT for the first time. Langmuir probe measurements suggest that the microwave heating maintains a bi-Maxwellian electron distribution, and that the plasma density decays on a millisecond time scale when heating ceases. Furthermore, a Langmuir probe mounted on an electronic moving stage measures profiles of plasma temperature and density with very high spatial resolution. These profiles profiles show evidence of magnetic islands, in agreement with electron-beam mapping of the vacuum magnetic field. Previously observed disagreements between Biot-Savart field calculations and experimental Poincaré maps indicated that some of the vacuum islands result from error fields [4]. Evidence of error fields was also provided by photogrammetric measurements of external coil misalignments [4]. We have now undertaken to diagnose error fields due to the internal coils by perturbing the calculations to reproduce the experimental results. We also present results of VMEC [5] free- and fixed-boundary calculations of CNT equilibria and ongoing work to upgrade the ECRH system from 1 to 16 kW.

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First W7-X simulations with the fast ion orbit following code ASCOT

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In stellarators, the confinement of fast ions is a key issue because their orbits are not intrinsically confined. The ASCOT code [1] has been used extensively to study the confinement of fast ions in tokamaks. Neutral beam injection (NBI) and fusion alpha particle simulations are routinely carried out for, e.g., ITER, JET and ASDEX Upgrade. This contribution describes how the code was modified to accommodate stellarator simulations. The goal of this work is to study the confinement of NBI ions in Wendelstein 7-X (W7-X).

ASCOT follows the orbits of markers that represent a large number of real particles. The generation of these markers for NBI simulations is done with the beamlet-based neutral beam ionisation code BBNBI [2]. The beam geometry of W7-X was recently included in the BBNBI library of injectors.

To follow the markers, ASCOT interpolates the magnetic fields using splines. In the tokamak version, the field consists of an equilibrium field calculated from the axisymmetric poloidal flux map and 3D perturbations from, e.g., the toroidal field coils. Support for three-dimensional equilibria was implemented for stellarator studies. The interaction of the markers with background plasma is modelled using Monte Carlo collision operators derived from the Fokker-Planck equation. The necessary plasma parameters are evaluated from 1D profiles using the 3D equilibrium.

In tokamaks, ASCOT is mainly used for simulating fast ion wall power loads and NBI heating and current drive. As the first stellarator result we will present W7-X wall power loads arising from NBI ions.

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Monte Carlo estimations of fast particle transport in quasi-symmetric devices

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The last fifty years of thermonuclear fusion research have allowed developing magnetic configurations with outstanding properties in terms of the confinement of particles and the stability of the plasmas they contain. Unfortunately, this doesn't mean that the necessary conditions for achieving controlled thermonuclear fusion have been reached yet, or that the underlying physical processes are fully understood. One of the unavoidable topics for the success of this research depends on understanding the mechanisms behind the confinement of fast particles. Moreover, fast particles are excellent probes to diagnose the quality of the magnetic field trap.

This work intends to simulate, analyze and extend our understanding about the particle and energy transport of fast particles depending on the symmetry of the magnetic field. To this end, Monte Carlo techniques are particularly attractive [1] since are able to simulate fast particle transport for arbitrary magnetic configurations without making any assumption on the nature of its transport, whether being diffusive, i.e. assuming a linear relation between fluxes and gradients, or even local.

The characterization and quantification of transport will be done by employing a whole set of tools, imported from the theory of stochastic transport, on the resulting particle trajectories. In this way, it will be possible to identify for which regimes and magnetic configurations non-diffusive dynamics maybe dominant, as well as to point to the relevant mechanisms responsible for it.

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Visible/IR Observations of the Poloidal Limiter in W7-X

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As part of the US/German collaboration on W7-X, we report on the design and preparation of a relay lens-based endoscope for infrared observations of the W7-X limiter and divertor components, during early operation. We are designing two high resolution systems, one to view the limiter in Operational Period 1.1 (OP1.1), and one to view the test divertor unit with ORNL scraper element in OP1.2. Optical access is obtained using a water-cooled, shutter-protected, reentrant viewing tube, which can be mounted in several possible diagnostic ports. It will contain 1.4 meter long borescope optics working at f/4 in the 3-5 micron wavelength band. Software for real-time acquisition and analysis written in Matlab with multi-core GPU image processing will also be discussed. For OP 1.1, we have prepared a high resolution view of the poloidal graphite limiter in W7-X for the first operational period in 2015. Magnetically shielded visible (400-700 nm) and mid-band infrared (3-5 micron) cameras share a nearly identical view through a large sapphire window mounted on the AEA30 port. Both systems achieve sub-mm spatial resolution and 10 millisecond time resolution while viewing three of the limiter tiles. We will compare heat flux patterns actually observed on the limiter with numerical predictions [1] corresponding to different plasma diffusivities.

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Isotope effect physics, turbulence and long-range correlation studies in tokamaks and stellarators

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The isotope dependence of plasma confinement is still one of the main scientific conundrums facing the magnetic fusion community after more than twenty years of intense research. Experimental studies have shown a reduction of the L-H power threshold by about 50% when using Deuterium or He instead of Hydrogen [1]. Based on present ITPA scaling laws, H-mode operation is expected to be marginally feasible in H while likely in He [2]. Thus, better understanding of the dependence of the L-H power threshold on isotope mass is urgently needed in order to improve ~~our~~ confidence in ITER scenarios.

In this paper, we present a study of the isotope effect from the prospective of multi-scale mechanisms. We have characterized turbulent radial scale structures and large coherent structures (e.g. GAMs, zonal flows) using probe measurement and correlation analysis in H and D plasmas in the TJ-II stellarator and in the ISTTOK tokamak. Experiments were performed in ISTTOK ohmic heated plasmas (B_T = 0.5 T, $I_p \approx 4 - 6$ kA) and in TJ-II Electron Cyclotron Resonance Heated (ECRH) low-density plasmas (B_T = 1 T, P_{ECRH} = 400 kW)

We have found that when changing plasma composition from ~~the~~ H to D dominated plasmas, the Long-Range-Correlation (LRC) amplitude increased (10-30%) while the particle confinement improved by around 50% in the ISTTOK tokamak, in agreement with previous findings in the TEXTOR [3] and FT-2 tokamaks [4]. In addition, both GAMs and low frequency zonal flows were found to contribute to LRC. On the contrary, the amplitude of LRC decreases slightly with ~~the~~ D/H ratio without any significant modification in ~~the~~ particle confinement in the TJ-II stellarator. The local radial correlation length (L_r) increases with the isotope mass both in ISTTOK and TJ-II in agreement with previous results [3, 6].

These findings show the impact of the isotope effect on both the largest scales (LRC determined by the device size) and the characteristic radial scale of turbulent structures. Differences in the level of isotope effect in tokamaks and stellarators could be a consequence of the stronger damping of zonal flows in non-optimized stellarators than in tokamaks.

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Interplay between long-scale length radial electric field components and zonal flow-like structures in the TJ-II stellarator

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The neoclassical non-ambipolar fluxes are formally dominant in determining the equilibrium (long scale length) radial electric field in stellarators [1]. Fluctuating, flux surface-constant zonal flows (ZF) have a specific interest, because they are thought to be instrumental in turbulent transport control and to be a natural state of turbulent plasmas [2]. A critical aspect for stellarator reactor development is clarifying the coupling mechanisms between turbulent transport and 3-D effects. For non-symmetric systems, it has been found that neoclassical optimization could lead to reduced ZF damping and, as a consequence, to turbulent transport optimization [3].

In this paper, we present a study of the interplay between long-scale length radial electric field components and zonal flow-like structures in the TJ-II stellarator. Experiments have been carried out in TJ-II NBI plasmas. A full set of plasma diagnostics has been used to characterize plasma parameters a unique detection system based on two rake Langmuir probe arrays located at two different toroidal locations [4]. The long-range-correlation (LRC) spectrum, computed from the toroidally separated potential measurements, is dominated by frequencies below 20 kHz.

Experiments in TJ-II have shown an empirical correlation between the amplitude of LRC (proxy of zonal flows) and the radial gradients of floating potential (proxy of radial electric fields) in NBI plasmas (L-mode regime). In other words, it shows an empirical coupling between neoclassical radial electric fields and the amplitude of Zonal Flow like structures. The amplitude of LRC strongly increases with increasing radial electric field reaching saturation for E_r values in the order of 1 kV/m. These results are qualitatively consistent with GK simulations showing the influence of radial electric fields in the zonal flow residual level [5, 6]. Interestingly, the radial structure of LRC, with a characteristic radial scale in the order of 1 – 3 cm, depends on plasma conditions. The observed interplay between neoclassical ExB shear flows and the development of low frequency zonal flow-like structures could be explained considering the role of electric fields as a turbulence symmetry-breaking mechanism or/and the influence of radial electric fields on particle orbits. These findings are consistent with previous experiments showing that the amplitude of LRC is amplified by external radial electric fields [4] as well as in the proximity of the L-H transition [7].

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On the nonlinear generation of zonal flows by turbulence in stellarators

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Zonal flows, here defined as $E \times B$ flows due to the flux-surface-averaged electrostatic potential, are an important element of turbulence in magnetic fusion devices, and have a reputation for a strong mitigating effect on the intensity of turbulence.

It is known that the linear behavior of these flows depends on the three dimensional shape of stellarator magnetic geometries. This is true of both the long-time (residual) behavior and the short time oscillations [1, 2]. However, very little is known about how magnetic geometry affects the nonlinear generation of zonal flows. In this work we investigate this question by analytic theory and direct numerical simulation.

We generalize the secondary instability theory of [3] to allow for arbitrary magnetic field geometry, and test its predictions directly against gyrokinetic simulations. Because the secondary mode growth rate depends on the root-mean-square amplitude of linearly unstable modes on a magnetic flux surface, we conclude that such modes are less effective at driving zonal flows when they are localized within a flux surface, and we present a series of nonlinear simulations demonstrating this effect.

As the space of stellarator configurations is large, absolute conclusion regarding the role of zonal flows in turbulence cannot be drawn. However, simulations show stellarator turbulence to be markedly localized [4]. Thus, from our results, it can be expected that Wendelstein-7X, and to some extent stellarators in general, should exhibit turbulence with less zonal flow activity than that found in typical axisymmetric configurations. However, we will argue that this fact need not imply stronger overall transport, and we propose that the feature may be exploited to actually lower transport.

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Radially local approximation of the drift kinetic equation in the conservative form

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Effects of neoclassical transport on plasma confinement are more significant in stellarator and heliotron plasmas than in tokamak plasmas because, in the former, radial drift motions of trapped particles in helical ripples enhance particle and heat transport due to nonaxisymmetry of the magnetic configuration. Conventional calculations of neoclassical transport fluxes are done applying radially local approximation to solving the drift kinetic equation, in which $\mathbf{v}_d \cdot \nabla f_1$ are often neglected as a small term of higher order in the normalized gyroradius parameter $\delta \sim \rho/L$. (Here, $\mathbf{v}_d$, f_1, ρ, and L represent the guiding center drift velocity, the deviation of the guiding center distribution function from the local Maxwellian equilibrium distribution, the gyroradius, and the equilibrium scale length, respectively.) However, in stellarator and heliotron plasmas, this $\mathbf{v}_d \cdot \nabla f_1$ term is known to be influential on the resultant neoclassical transport because it significantly change orbits of particles trapped in helical ripples. Therefore, at least, the $\mathbf{E} \times \mathbf{B}$ drift part $\mathbf{v}_E \cdot \nabla f_1$ in $\mathbf{v}_d \cdot \nabla f_1$ has been kept in most studies of neoclassical transport in helical systems. Recently, it was shown by Matsuoka *et al.*[1] that the neoclassical transport is significantly influenced by retaining the magnetic drift tangential to flux surfaces in $\mathbf{v}_d \cdot \nabla f_1$ for the magnetic configuration of LHD especially when the radial electric field is weak. However, as pointed by Landreman *et al.*[2], stationary solutions of the drift kinetic equation with the radially local approximation used require additional artificial sources (or sinks) of particles and energy when the above-mentioned drift terms are retained. In this work, a novel radially local approximation of the drift kinetic equation is presented. The new drift kinetic equation that includes both $\mathbf{E} \times \mathbf{B}$ and tangential magnetic drift terms is written in the conservative form and it has favorable properties for numerical simulation that any additional terms for particle and energy sources are unnecessary for obtaining stationary solutions. Besides, it is shown that the intrinsic ambipolarity condition for neoclassical particle fluxes are satisfied in the presence of quasisymmetry of the magnetic field strength.

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SFINCS: A flexible tool for advanced stellarator neoclassical computations

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SFINCS is a versatile continuum code for solving several variants of neoclassical kinetic problems in nonaxisymmetric plasmas [1,2]. Coupled 4d (poloidal angle, toroidal angle, pitch angle, speed) kinetic equations for each species are solved, and if desired, the quasineutrality equation can simultaneously be solved at each point on the flux surface. Parallelized solution of the 4d problem is accelerated with an iterative Krylov method, preconditioned with a reduced 3d problem. The full linearized Fokker-Planck collision operator is implemented between each pair of species, with no approximation of the field term or expansion in mass ratio. Efficient representation of velocity space is achieved using a pseudospectral method based upon non-classical orthogonal polynomials [3]. Optional nonlinear terms in the kinetic equation (involving both the non-Maxwellian distribution function and poloidal/toroidal electric field) can be included using Newton's method. A variety of models for terms involving E_r are available to allow comparison between models. The numerical problem is formulated to permit solution of a wide variety of kinetic equations, whether or not phase space volume and/or energy are conserved, so individual terms can be turned on or off to examine their effect. Several applications will be presented [4,5].

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Calculation of the toroidal torque due to non-axisymmetric magnetic field perturbations in a tokamak with the SFINCS code

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The toroidal plasma rotation in tokamaks is influenced by the toroidal torque due to non-axisymmetric magnetic field perturbations (neoclassical toroidal viscosity). We determine this torque with the Sfincs code [1], which solves the 4D (poloidal angle, toroidal angle, pitch angle, speed) drift kinetic equation in general 3D equilibria for several particle species using the full linearized Fokker Planck collision operator, see the contribution by M. Landreman *et al.* to this conference. The conventional method to calculate the torque is to obtain it indirectly from the radial particle fluxes via the flux-force relation. Alternatively, one can determine the toroidal torque directly from the relevant moment of the distribution function. These two approaches are compared here in terms of their convergence properties in the numerical resolution parameters in the four dimensions of the distribution function in Sfincs. The computing time and processor memory requirements increase dramatically at low collisionality, so it is important to use a method which does not demand too high resolution for convergence. Benchmarks against the NEO-2 code [2] are also performed.

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Effect of magnetic drift tangential to flux surface on local neoclassical transport in non-axisymmetric plasmas

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In evaluating neoclassical transport by radially-local simulations, the magnetic drift tangential to a flux surface is usually ignored in order to keep the phase-space volume conservation. However, validity and an effect of the approximation have been never verified due to the difficulty to numerically retain the drift. In the present work, we provide a novel formulation for a drift kinetic equation, in which the compressibility by the tangential magnetic drift is regarded as a source term. The formulation enables ones to solve the drift kinetic equation by a two-weight δf Monte Carlo method for non-Hamiltonian system [1]. Based on these approaches, a new radially-local neoclassical transport code is developed.

To address the effect of the tangential drift on the local neoclassical transport, we classify various neoclassical transport models into four models based on different approximations used in the models; (1) radially-global model including a finite orbit width (FOW), (2) zero orbit width (ZOW) approximation which ignores the radial drift but retains all other drifts such as the tangential magnetic drift, (3) zero magnetic drift (ZMD) limit, which excludes the tangential drift from the ZOW limit, (4) DKES-like limit, which is conventional and widely adopted in existing codes such as DKES, GSRAKE, *etc*.

A systematic comparison of these neoclassical transport models is performed using the new code. Figure 1 shows the particle flux dependence on the radial electric field (E_r) for the various models. The ZOW model is demonstrated to substantially changes the dependence. The peaked behavior of the local neoclassical fluxes around $E_r = 0$ is removed by taking the tangential magnetic drift into account while staying within a radially-local model.

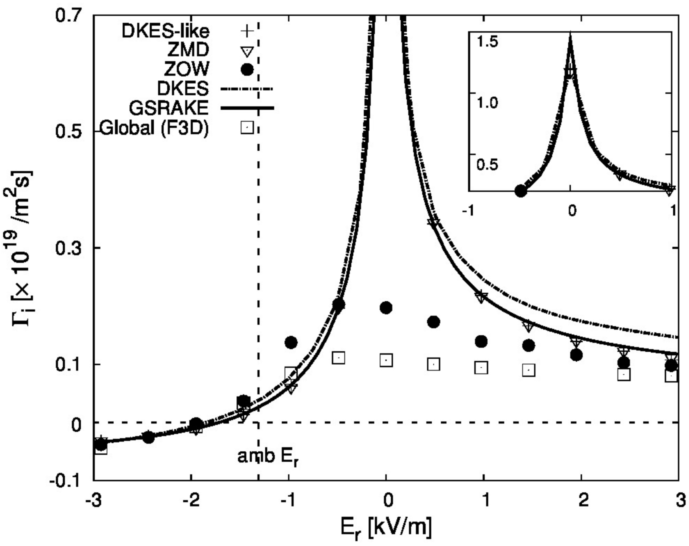

Fig.1 Neoclassical particle flux dependence on the raidal electric field. DKES-like, ZMD, and ZOW represent the local neoclassical transport models described in this abstract. Results obtained by DKES, GSRAKE, and FORTEC-3D codes are also plotted, where FORTEC-3D is denoted as Global (F3D). The ambipolar E_r predicted by GSRAKE is shown by a vertical line for a reference purpose. Shown in upper right is an enlarged view.

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Comparison of bootstrap current calculation in helical plasmas among different types of approximations in drift-kinetic equation

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In toroidal plasma, neoclassical (NC) theory describes radial NC flux and the bootstrap current. The bootstrap current depends on Magnetohydrodynamics (MHD) equilibrium and collisionality. The bootstrap current is unessential for MHD equilibria of helical plasma. However, in high-beta and high-temperature plasma, the bootstrap current could be large enough to affect equilibrium field. Therefore, a self-consistent method is required to track MHD equilibrium and bootstrap current. For example, Wendelstein 7-X (W7-X) is optimized to have good MHD instabilities and low bootstrap current in order to improve NC confinement.[1] Here W7-X design fellowed the local NC code.[2] Conventionally, mono-energy and local approximation are employed to reduce computation time and resource. Here, "local" means neglecting the higher order magnetic drift terms in the drift-kinetic equation. The recent NC studies[3][4] indicate that conventional local approximations are deficient for some conditions in radial neoclassical transport. It shows that keeping tangential magnetic drift terms leads to large differences in the radial NC flux where $\mathbf{E} \times \mathbf{B}$ rotation is small. The tangential magnetic drift effect on bootstrap current is still an open question. This work performs the benchmark between conventional and new local code with the tangential magnetic drift and discusses its contribution on bootstrap current in LHD and W7-X.

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Calculations of Bootstrap and Pfirsch-Schlüter currents in stellarator geometry

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Bootstrap and Pfirsch-Schlüter currents are internal plasma currents driven by the pressure gradient. These currents can influence the rotational transform with potentially negative consequences for the divertor operation and plasma confinement. Therefore, it is important to carefully assess them. In our contribution, we consider dependence of the bootstrap current on the collisionality in stellarator geometry. The Wiener-Hopf problem is analytically addressed taking into account the stellarator-specific radial drifts. It is shown that the collisionality correction to the bootstrap current arises not only in the trapped-passing boundary layer, as is the case for tokamaks, but also in the trapped-trapped boundary layers. In the second part of the presentation, the Pfirsch-Schlüter current is numerically evaluated in stellarator geometry solving the fluid equations as an initial value problem. The numerical procedure is validated in tokamak geometry where comparisons with an analytic expression is available. In stellarators, it is demonstrated that singular currents can develop at the resonant flux surfaces.

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Magnetic islands, bootstrap current and 3D MHD modeling of W7-X

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Recent work on the impact of plasma profiles and bootstrap currents on the magnetic structure in W7-X is summarized and initial work on developing the capability to use the 3D extended MHD HiFi code to model W7-X plasmas is presented.

The impact of a range of density, electron temperature, and ion temperature profiles on self-consistent bootstrap current, ballooning stability and the magnetic structure of equilibria in computational studies of the W7-X stellarator is examined using the VMEC [1] and SIESTA [2] codes. Previous work has shown that even a small bootstrap current can change the rotational transform profile and thus, change the magnetic configuration, especially in the edge region. In this work, free-boundary equilibria for the W7-X coil configuration have been obtained over a range of pressure, density and temperature profiles, including equilibria with self-consistent bootstrap current (i.e., where the plasma current is solely from the bootstrap current). The impact of these profiles on bootstrap current, magnetic structure in the edge, and ballooning stability is examined. The formation of islands in the edge regions and correlation with ballooning stability is discussed. Methods of ameliorating the impact of bootstrap current are also discussed.

Work is underway to develop the capability to use the 3D extended MHD HiFi [3] modeling framework to simulate plasma evolution in W7X, utilizing VMEC to produce initial conditions. HiFi offers high order numerical accuracy, tractable code structure, flexible specification of the physics to be considered, and fully three-dimensional capabilities. This work includes obtaining appropriate VMEC equilibria that reflect the early stages of a particular shot. Development of an interface is underway to then use a VMEC equilibrium as the initial condition in HiFi. Once the implementation is complete, verification of HiFi simulations of W7-X plasmas will begin.

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Commissioning and Test of the ECRH system of Wendelstein7-X

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The standard heating method at W7-X, either for plasma start-up or bulk heating, will be electron cyclotron resonance heating (ECRH) by means of ten 140 GHz Gyrotrons with 1 MW nominal power each. The ECRH-system is also the only heating system which is able to operate steady state. The quasi-optical transmission line to the plasma vessel is the first of its kind. It consists of ten single beam sections (SBS) matching the non-perfect Gaussian beam output of the gyrotrons to the subsequent multi beam section (MBS) allowing low-loss transmission with an overall spurious mode generation of less than 0.2 %. The whole MBS consists of two lines, each with 4 focusing and 3 plane large-area multi-beam mirrors transmitting six beams arranged in a circle around a central beam. The overlap of the beams on the mirrors reduces the mirror surface and leads, important for the envisaged 30 min operation, to a more homogenous heat load over the mirror surface caused by ohmic losses of about 0.2 % per mirror. The quasi-optical transmission is intrinsically broadband. Finally, the MBS is followed by a further single beam section dividing the beams to the particular vacuum window of the four equatorial launchers installed on W7-X with three beam lines each. The plasma facing mirror of each beam line is steerable in poloidal and toroidal direction to vary the radial heat deposition within the plasma and to drive additional plasma current, respectively. All mirrors are sufficiently water-cooled for steady state operation. The mirrors of the whole transmission line were aligned with the aid of a cross-line laser and the high power microwave beam itself of the chosen six gyrotrons for first operation phase of W7-X (OP1.1). Only two beams were necessary for the final adjustment of the MBS mirrors demonstrating the nearly perfect imaging properties of the imaging MBS. The overall power loss up to the launcher window is 5 % including diffraction, beam truncation, misalignment, absorption of the mirrors and the atmosphere. In addition, 2 % power loss is expected by the launcher. In the first operational phase six gyrotrons with a total power of about 5 MW will be available for the commissioning of the machine. During their alignment the beam position and polarisation were measured by the transmission diagnostic (ECA) at the heat shield opposite to the launcher. The stray radiation level for the case no absorption could also be determined by so called ECRH-sniffer probes in every of the 5 W7-X modules. The system is ready for operation and first plasma start-up studies. Plasma operation is expected in autumn this year.

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First results from protective ECRH diagnostics in W7-X

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Wendelstein 7-X is a fully optimised stellarator designed for steady state operation. The main heating system is electron cyclotron resonance heating (ECRH) launching microwaves at 140GHz into the vessel. The ECRH system can provide up to 10MW microwave power for up to 30 minutes which is equivalent to steady state operation from the physics point of view. Non absorbed ECRH power can easily lead to overheating of in vessel components, especially in long discharges. A lot of effort has been spent on the protection of components from thermal loads during the designs phase of W7-X. However, this passive protection is only efficient up to a certain level. Therefore, a set of diagnostics has been developed to protect the machine from non absorbed ECRH power. The power is launched into the machine by remote steerable quasi-optical launchers from the low field side in X- or O-mode. While in X-mode the first pass absorption is rather high ($\approx 99\%$), it is only $40 \ldots 60\%$ in O-mode heating scenarios. The non absorbed power hits the inner wall where it can be measured by waveguides embedded in the tiles covering the wall [1]. In order to prevent the inner wall from overheating a near-infra red sensitive video diagnostic has been integrated in the ECRH launchers [2]. The dynamic range of the video diagnostic is between 450 and 1200°C which is well within the specified safe operating range of the inner wall tiles. Thermal calculations predict a temperature increase above the detection threshold of the video system for scenarios of plasma start-up failure or poor absorption on a time scale of $\approx$ 100ms and the risk of overheating after $\approx$ 300ms. These predictions are compared to first experimental results. The stray radiation level inside the machine is measured by so called sniffer probes which collect all radiation approaching the probing surface independent of their propagation angle and polarisation. Five sniffer probes are installed at different toroidal positions allowing a comparison of the measured stray radiation distribution to model predictions.

The sniffer probes are integrated in the interlock loop of the ECRH system in order to switch of the heating in case of increased stray radiation levels. For the future this is also planned for the the video diagnostic which ha no real-time capabilities at the moment.

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3D Magnetic Field Effect on ECRH/ECCD in Helical Systems

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Launching conditions for electron cyclotron resonance heating (ECRH) and current drive (ECCD) should be optimized for efficient heating and current drive and for suppression of non-absorbed waves which may cause damage of in-vessel components. The polarization of launched waves is controlled in order to make single pass absorption maximal. In a uniform magnetic field, the ordinary (O) and extraordinary (X) modes propagate independently due to the difference in refractive index, while in a non-uniform 3D magnetic field with finite magnetic shear, they are coupled, changing the power fraction along propagation path. This phenomenon was theoretically predicted by using wave coupled equations, and experimentally clarified in the Heliotron E helical device which has a strong magnetic shear [1]. Since the LHD device has a 3D non-uniformity likely as the Heliotron E device, the mode coupling should be taken into account for the optimization of launching condition. On the other hand, the mode coupling effect is expected to be weak in the Heliotron J device which has an weak magnetic shear. A vector equation using Stokes parameter [2][3] is solved to calculate coupling between the O- and X-modes in the Heliotron J configurations. Calculation results in the standard Heliotron J configuration are in good agreement with the polarization dependence of experimental single pass absorption rate. The calculation will also be performed for the LHD configuration, and compared with the experimental results. Effect of edge plasmas outside the last closed flux surface will be discussed.

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Self-consistent ICRH modelling in Wendelstein 7-X plasmas

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Ion Cyclotron Resonance Heating (ICRH) will represent a substantial heating source in future Wendelstein 7-X (W7X) operation phases. Simulations of ICRH scenarios are required in order to understand and predict the wave deposition and the fast particle distributions in W7X. The SCENIC package [1] is applied to minority ICRH scenarios in W7X, which computes MHD equilibrium (ANIMEC) [2], heat deposition at the fundamental frequency on minority species (LEMan) [3-4] and the associated fast ion distribution function (VENUS-LEVIS) [5]. We present SCENIC simulations of a 4He plasma with a 1% H minority and an antenna model close to the design foreseen for the W7X ICRF system. We compare a high mirror with a standard equilibrium and discuss the implications of the magnetic equilibrium configuration on the wave deposition and the fast ion distribution function. These calculations assume an isotropic and thermal minority ion distribution. It is found that the antenna localisation breaks the five-fold periodicity of the equilibrium. We assess the heat transfer through the toroidal periods via Coulomb collisions between the minority ions and the background electrons and ions. Further simulations will provide improved calculations of the wave field and the saturated fast ion distribution function, in particular yielding more consistent power depositions on each toroidal segment of the machine. On-going code developments will resolve the full-orbit particle trajectories on both sides of the last closed flux surface [6], allowing the calculation of particle fluxes on the walls. One important extension to the LEMan heating code will entail the assessment of ICRH with a three-ion species scenario [7].

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Recent Progress of High-beta Experiments in LHD

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In previous LHD experiments, the achieved volume averaged beta value has been increased by optimizing magnetic configurations to characteristics of equilibrium, stability and transport with an increase in heating power. The beta value of over 5 % has been obtained in B_t < 0.5 T configurations [1]. Recently, we have made experiments at B_t =1 T in order to produce high-beta plasma in low-collisional regime. The R_{ax} scan experiments in the standard configuration (pitch parameter of helical coil, γ_c = 1.254) with constant electron density suggest that the optimum R_{ax} is 3.56 m for production of high beta plasma. In the experiments, the beta value of 4.1 % was achieved by injecting four Hydrogen pellets and maintained for about 0.1 s. In the pellet case, peaked density profile was formed, which induced large Shafranov shift (ΔR ~ 0.44 m). Then strong m/n = 2/1 mode appeared because of formation of steep pressure gradient around the $\iota/2\pi$ = 1/2 resonant surface. In the case of gas-puff discharge, the achieved beta value was 3.4 % and maintained for an enough long time. Figure 1 shows the typical discharge produced by the Hydrogen gas-puff. The plasma was produced at 3.3 s by three tangential NBIs and the beta value was increased at 3.7 s by additional heating due to two perpendicular NBIs and ICRF. The beta value was spontaneously increased at 3.75 s with an increase in electron density, mainly, in peripheral region. The ion saturation current measured at the divertor plate was significantly reduced then, which means that the particle confinement in the periphery was improved although no clear reduction of Hα was observed. After the transition, edge MHD instabilities such as m/n = 1/2, 2/3 and so on appeared and were enhanced by formation of the steep pressure gradient in the periphery, which obviously limits the increase in the beta value. The HINT2 calculations before and after the transition suggest that the increase in the beta value is due to extension of the region with long connection length of magnetic field line to the outward.

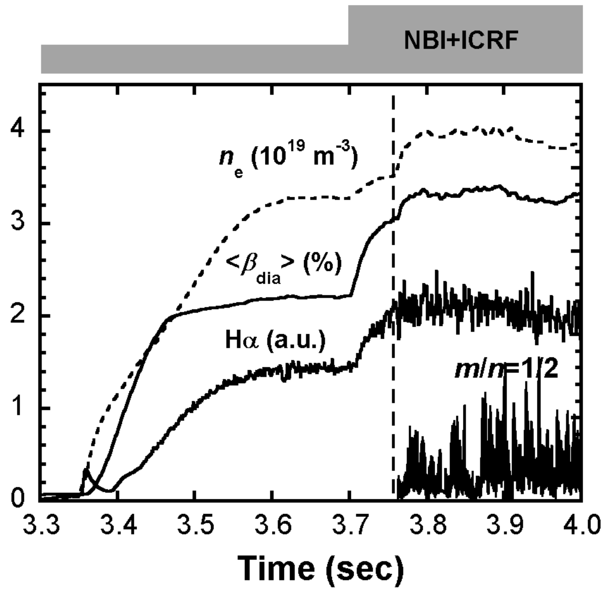

Fig.1 Typical high-beta discharge produced by Hydrogen gas-puff

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Spontaneous healing of magnetic islands in the LHD by plasma flow

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Magnetic islands present in the vacuum configuration of LHD plasmas can be spontaneously healed. Recent experimental and theoretical works suggest that the properties of the plasma flow [1-7] have a powerful influence on the magnetic island behavior. In a previous LHD experiment [8], the magnetic island states (growth or healing) can be clearly divided into two regions in the β and the collisionality space. While the β and the collisionality could correlate with island physics through Pfirsch-Schlüter (PS) and bootstrap (BS) current effects, efforts to understand these results via these mechanisms demonstrated that other ingredients, unknown at that time, were necessary to explain island behavior. The understanding of the decision mechanism of that boundary dividing the island states had been left as a problem to solve. A recent study has found that the dynamics of the magnetic island is affected by the poloidal plasma rotation [3], which shows the experimental fact that the poloidal flow changes prior to the transition of the magnetic island from growing to healing. The poloidal flow hysteresis is also observed in the magnetic island transition dynamics: when the magnetic island is suppressed once by the high poloidal flow the suppression lasts until the poloidal flow becomes small enough. Associated with the transition from a locked island to a healed state is a large change in the rotation profile in the vicinity of the rational surface. Focusing on the poloidal flow, theories based on the torque balance relations have been proposed to describe transitions between two asymptotic states [2, 4, 5-7]. Figure 1 shows the relationship between the theoretically derived critical beta [4] and experimentally obtained beta, which are consistent with empirically derived scaling. Comparisons between the theory and observations on LHD show favorable agreement.

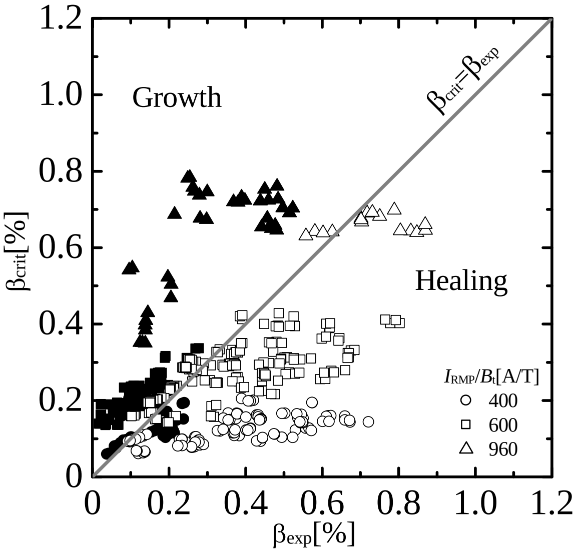

Fig. 1 Island healing transition boundary. Closed and open symbols indicate island growth and healing, respectively. Normalized RMP coil current I_{RMP}/B_t are 400A/T (circle), 600A/T (square), and 960A/T (triangle), respectively.

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Implementation of a Coherence Imaging Diagnostic for the Compact Toroidal Hybrid

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An optical coherence imaging diagnostic is being commissioned for time-resolved measurements ($\sim$ 10 ms) of ion emissivity, velocity, and temperature in the Compact Toroidal Hybrid (CTH) experiment. The coherence imaging technique[1] measures the spectral coherence of an emission line with an imaging interferometer of fixed delay. Coherence imaging has a number of advantages when compared to dispersive Doppler spectroscopy, including higher throughput and the capability to provide 2D spectral images, making it advantageous for investigating the non-axisymmetric geometry of CTH plasmas. Furthermore, detailed measurements of the ion flow structure provided by coherence imaging combined with predictive computational models have been used in tokamak divertor regions to investigate particle and heat transport[2, 3] and could also provide spatially resolved images of the complex flow structure associated with the island divertor. The coherence imaging technique can also be extended to yield the orientation and magnitude of the magnetic field by measuring the polarized spectral components due to Zeeman splitting. A spectral survey of the visible and ultraviolet emission for a range of CTH discharges has been conducted and identified possible helium and carbon impurity lines that could be utilized for coherence imaging measurements in CTH. Results from this diagnostic will aid in characterizing the equilibrium ion parameters in both the edge and core of CTH plasmas for planned island divertor and MHD mode-locking experiments. Initial interferograms from a calibration light source and CTH plasmas will be presented.

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Doppler Coherence Imaging of Ion Dynamics in VINETA II

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In magnetically confining plasma experiments, diagnosis of ion flows is of great importance to measure the plasma response to the magnetic field or the exhaust particle flows in the divertor areas. Doppler coherence imaging spectroscopy (CIS) is a relatively new technique for the observation of plasma bulk ion dynamics [1]. It is a passive optical diagnostic enabling line-integrated measurements to obtain 2D images of the ion flow and ion temperature. The general principle is similar to traditional Doppler spectroscopy, however CIS uses an imaging interferometer to perform narrow-bandwidth Fourier spectroscopy.

A major advantage of the coherence imaging technique is the large amount of spatial information recovered. This allows tomographic inversion of the line-integrated measurements. With existing CIS setups, scrape-off-layer and high field side edge impurity flows could be observed in the MAST [2], core and edge poloidal He II flows in the WEGA stellarator [3] and divertor impurity flows in DIII-D [4].

The main objective of this study is the research of ion dynamics with the CIS during a driven magnetic reconnection event in the linear plasma experiment VINETA II. VINETA II is dedicated to the study of both collisional and near collisionless driven reconnection [5]. Magnetic reconnection is of importance in space phenomena, such as solar flares or in the Earth's magnetosphere, as well as in magnetic confinement fusion experiments. To measure the ion dynamics during a magnetic reconnection event, measurements are performed in VINETA II due to its high discharge reproducibility. It will be investigated whether magnetic reconnection has an influence on VINETA ion dynamics. Therefore, CIS flow resolution needs to be improved as well as temporal resolution. The CIS diagnostic is currently being set up and first flow measurements without reconnection will be presented.

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Recent development of 2D potential measurement with heavy ion beam probe on the Large Helical Device

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Heavy Ion Beam Probe (HIBP) is a very useful diagnostic tool to study the radial electric field formation physics in torus magnetic confinement devices. With this diagnostic tool, we can measure, simultaneously, equilibrium and fluctuation of electrostatic potential and density in the core region of high temperature plasmas. For this unique advantage, many HIBP systems are installed on a variety of magnetic confinement devices, such as tokamaks, reversed field pinches, mirror devices, and helical devices.

In the Large Helical Device (LHD), an HIBP was installed and has been developed to study physics related to the radial electric field in a helical confinement plasma [1-2]. In HIBP, one dimensional profile of potential is usually measured by changing the incident angle of probe beam. If the probe beam energy is changed additionally, the observation position can be changed two-dimensionally and the two-dimensional potential profile can be obtained. However, the beam transport line of LHD-HIBP system is long, ~ 20 m, therefore, the adjustment of beam (position and focusing) on this line is needed when the beam energy is changed. To reduce the required time for this adjustment, an automatic beam control system on the beam line was developed [3]. By using this system, the alignment of the beam on the beam line can be completed during a typical discharge cycle (three minutes). A two-dimensional equilibrium electrostatic potential profile was successfully measured in LHD by changing the probe beam energy shot-to-shot. In a recent experiment, measurement of the potential structure in islands produced by rmp coils was tried. Definite potential structure in islands was not able to be measured in this experiment, but a flat potential profile in one-dimension was obtained. Details of the automatic beam control system and recent experimental results will be reported.

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Generation of rotational transform in a tilted-coil solenoid-free "tokamak"

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Experimental evidence was obtained, by means of an electron beam, that rotational transform can be generated in a toroidal configuration constructively similar to a tokamak, but solenoid-free and featuring six tilted toroidal-field coils. The coils are planar and, in fact, circular, hereby the device name CIRCUS [1]. In addition, the coils are interlinked to each other, which helps reducing the aspect ratio but is not strictly required. No current drive is necessary either, as theory and experiment confirmed. Comparisons between calculations and field-line mapping measurements for various coil tilts and positions will be presented, as well as predictions for devices featuring more coils, resulting in more axisymmetric plasmas. These are expected to operate at lower plasma current than a tokamak of comparable size and magnetic field, which might have interesting implications for disruptions and steady-state operation. Additionally, the toroidal magnetic ripple is less pronounced than in an equivalent tokamak in which the coils are not tilted.

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Optimization of heliotron-type magnetic configuration with modular coils and helical coils

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The optimization of magnetic field configuration of LHD was made based on the Fourier modes of boundary shape [1] with the fixed boundary calculation of equilibria using the VMEC code. This new configuration has a reduced Shafranov shift for high beta equilibria which greatly mitigates the degradation of the neo-classical confinement of LHD type heliotron configuration for high beta discharges [2]. The feasibility of the experimental device for the optimized configuration was examined by designing the helical coils using the STELLOPT code and a simple solution of the helical coil shape was found for the new configuration [3]. This success is unique because the standard method of designing magnetic coils for the optimized configuration is the modular coil design through the current density calculation on the current carrying surface. As well as the similar helical coil design to LHD device, the modular coils were also designed for the new configuration and a nice solution was obtained. The motivation of the magnetic coil design for both concepts is to investigate the divertor magnetic field structures. Another motivation is to evaluate the spacing between the plasma boundary and the magnetic coils because this value is the most important design parameter of the fusion reactor. For these aspects of stellarator configuration design, the comparison of modular coils and continuous helical coils will be given.

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Recent Advances in Stellarator Optimization

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Computational optimization has revolutionized the field of stellarator design. To date, optimizations have focused primarily on optimization of neoclassical confinement and ideal MHD stability, although limited optimization of other parameters has also been performed. One of the criticisms that has been levelled at this method of design is the complexity of the resultant field coils. Recently, a new coil optimization code - COILOPT++, which uses a spline instead of a Fourier representation of the coils, - was written and included in the STELLOPT suite of codes. The advantage of this method is that it allows the addition of real space constraints on the locations of the coils. The code been tested by generating coil designs for optimized quasi-axisymmetric stellarator plasma configurations of different aspect ratios. As an initial exercise, a constraint that the windings be vertical was placed on large major radius half of the non-planar coils. Further constraints were also imposed that guaranteed that sector blanket modules could be removed from between the coils, enabling a sector maintenance scheme. Results of this exercise will be presented. New ideas on methods for the optimization of turbulent transport have garnered much attention since these methods have led to design concepts that are calculated to have reduced turbulent heat loss. We have explored possibilities for generating an experimental database to test whether the reduction in transport that is predicted is consistent with experimental observations. To this end, a series of equilibria that can be made in the now latent QUASAR experiment have been identified that will test the predicted transport scalings. Fast particle confinement studies aimed at developing a generalized optimization algorithm will also be discussed.

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Approximate Quasi-Isodynamicity at Finite Aspect Ratio in a Stellarator Vacuum Magnetic Field

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A stellarator vacuum field is found in which, at finite aspect ratio, the contours of the second adiabatic invariant of nearly all particles reflected inside that surface are poloidally closed.

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Mini-Stellarator for public outreach

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Stellarators feature prominently in fusion research, but the plasmas produced in the big machines — and often the even machines themselves — are not directly visible for most visitors to the institutes, where they are operated. To improve this situation, an experiment called "Mini-Stellarator" is being built at Max-Planck-Institute for Plasma Physics (IPP) in Greifswald. This device can be controlled directly by visitors to IPP; it can produce a visible plasma and can demonstrate magnetic confinement of plasma.

The magnetic field for plasma confinement is created by two pairs of coils: two parallel coils, outside the vacuum chamber and two interlocking, water-cooled coils inside the vacuum chamber. Each pair of coils has a separate power supply and both power supplies can be manipulated by the visitor, allowing him/her to search for the configuration in which the magnetic field confines the plasma. The design of the Mini-Stellarator is based on the Columbia Non-Neutral Torus (CNT), which used a similar coil arrangement to confine pure electron plasmas. Plasma is created when the electron beam from an electron gun ionizes background gas; the background pressure is typically in the 10^{-3} mbar range.

The device is table-top size (approximated 1m high) with an inner coil diameter of 14 cm and a vacuum chamber volume of 90 liter.

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On the origin of negative current induced axially symmetric oscillations detected in L-2M stellarator experiments

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Nowadays, it is widely believed that negative current induced magnetic surface with rotational transform μ =0 in toroidal devices splits due to the action of equilibrium plasma induced magnetic fields (see, e.g., [1, 2]). Here, definition “negative” stands for the current directed in such a way as to reduce rotational transform μ ($\mu \equiv 1/q$, where q is the safety factor). There is another possibility for $\mu = 0$ surface splitting, namely, the action of axisymmetric (AS) tearing modes (see, e.g., [3]). Therefore, axisymmetric oscillations (ASO) can appear either due to quasi equilibrium splitting of $\mu = 0$ magnetic surface or due to the action of tearing modes, or combination of these two reasons. Stellarator experiments where negative current was large enough to create magnetic surface with $\mu = 0$ inside plasmas are well known (see, e.g., [4-6]). However, ASO geometrical structure remains uninvestigated.

Stellarator L-2M is intrinsically non symmetric system. Axisymmetric oscillations (ASO) with $m \neq 0$ (here m is the poloidal mode number) were observed at specific conditions, namely, when the sufficiently large (in absolute value) negative current is induced in electron cyclotron heated (ECH) plasmas. The aim of this work is to determine ASO geometrical properties and origin. For this purpose and have performed experiments and analyze them with estimations and calculations.

Experiments were performed at 5 kA $\leq |J| \leq$ 16 kA. At sufficiently small negative currents, i.e. $|J| \leq 7$ kA, an average value of magnetic fluctuations diminishes moderately, compared to zero net current plasma. ASO with $m = 2$ are observed at $|J| \geq 9$ kA, however at $|J| \leq 10$ kA they are observed in individual cases among considered and at $|J| > 10$ kA arise in each case. It is shown that necessary condition of ASO with $m = 2$ appearance is the compensation of the dipole components of magnetic fields produced by negative net current, Pfirsch-Schlűter current and that of the “effective” average magnetic field at the magnetic surface with $\mu \approx 0$. In addition it is argued that AS tearing modes do not participate in ASO initiation.

Threshold for ASO appearance can be calculated with reasonable accuracy within classical approach. However, peculiarities of current penetration into plasmas demonstrate that non classical effects are significant.

In zero net current plasmas ($\mu \equiv \mu^*$, where μ^* is the vacuum rotational transform) the central part of plasma is stabilized due to the effect of self stabilization. Negative current at $\mu < 0$ and $\mu^* > 0$ transforms stellarator self stabilization to self destabilization. It is shown that at sufficiently large in absolute value negative current instability zone (with respect to ideal and resistive interchanges) is formed in the central part of plasma and, hence, can be the reason for the observed degradation of plasma confinement. Changes in topology of magnetic surfaces are estimated. Arguments are adduced in favor of the hypothesis that current density profile is dependent on $|J|$. Distinctions in negative current effects between tokamak and stellarator are briefly outlined.

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Effects of Weak Pressure Gradients along Magnetic Field Lines, and of Stellarator Symmetry, in Plasma Equilibria with Magnetic Islands

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In toroidal plasma equilibria, cross-field transport is sufficiently strong relative to parallel transport in the neighborhood of small magnetic islands, and in the neighborhood of the X-lines of larger magnetic islands, that a small, but significant, pressure gradient along the field lines is maintained. There is a significant variation of the pressure within the flux surfaces. One consequence is that pressure driven currents near the X-lines are significantly affected by "stellarator symmetry", that is, by invariance under combined reflection in the poloidal and toroidal angles. In non-stellarator-symmetric configurations, the resonant Pfirsch-Schlüter (PS) currents have logarithmic singularites at the X-lines. In stellarator-symmetric configurations, the singular component of the current vanishes to leading order. The logarithmic singularity of the PS current is to be contrasted with equilibria having $\mathbf{B} \cdot \nabla p = 0$, where the singular component of the current is absent. It is also to be contrasted with 3D equilibria having simply nested flux surfaces, where the current density goes like $1/x$ near rational surfaces, and where x is the distance from the rational surface. The MHD equilibrium equations in their conventional form cannot be used to study these effects because they imply $\mathbf{B} \cdot \nabla p = 0$. We work with a closed, self-consistent set of MHD equilibrium equations that involves only perpendicular force balance, and is decoupled from parallel force balance. Proper treatment of the parallel component of force balance requires the retention of terms not included in the conventional MHD equations. The resulting equation determines plasma flow along the field lines, and is one of the transport equations rather than one of the equilibrium equations. The equilibrium equations are first applied to an analytically tractable magnetic field with an island to obtain explicit expressions for the rotational transform and magnetic coordinates, and for the pressure-driven current and its limiting behavior near the X-line. A more general calculation is then done of the pressure-driven current near an X-line and of the rotational transform near a separatrix. The study presented here is motivated, in part, by tokamak experiments with nonaxisymmetric magnetic perturbations, where significant differences are observed between the behavior of stellarator-symmetric and non-stellarator-symmetric configurations with regard to stabilization of edge localized modes (ELMs) by resonant magnetic perturbations (RMPs). Implications for the coupling between neoclassical tearing modes (NTMs), and for magnetic island stability calculations, are also discussed.

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Session 2

Edge-core coupling of turbulence and transport

Non-local interaction of ion thermal transport between core and edge triggered by ECRH on the LHD plasma

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The edge-core-related dynamics of the ion thermal transport during the spontaneous formation of ion ITB has been studied in the LHD [1]. Recently non-local interaction of ion thermal transport caused by ECRH was observed. Figure 1 shows the time evolution of (a) the port through power of NBI and ECRH, (b), (c) the gradient of T_e and T_i at the normalized minor radius of r_{eff}/a_{99} = 0.31 and 0.98, and the radial profiles of (d) T_e and (e) T_i before (4.14 s) and during ECRH superposition (4.74 s, P_{ECRH} = 5.1 MW). On-axis ECRH was superposed in stepwise up to 5.1 MW and the ray trace calculation showed that more than 90% of ECRH absorption power was deposited inside r_{eff}/a_{99} = 0.2 at 4.74 s. Both in the plasma core and the edge, T_e and the gradient increased with increase of ECRH power. On the other hand, the gradient of T_i was degraded in the plasma core with increase of the ECRH power; in contrast the T_i gradient was improved at the plasma edge. The normalized ion thermal diffusivity $\chi_i/T_i^{1.5}$ at r_{eff}/a_{99} = 0.98 reduced by 60%. The improvement of the ion thermal transport at the edge led to increase of T_i in whole plasma region even the core transport was degraded. Also the behaviour of the turbulence will be discussed in the presentation.

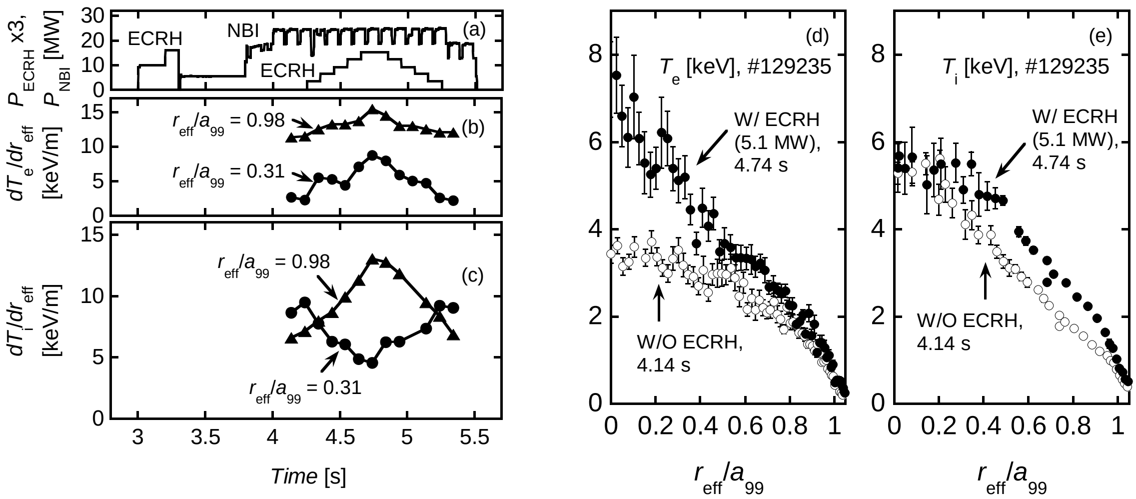

Fig. 1. The time evolution of (a) NBI and ECRH power, (b), (c) the gradient of T_e and T_i at r_{eff}/a_{99} = 0.31 and 0.98, and the radial profiles of (d) T_e and (e) T_i before and during ECRH superposition.

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Evolution of the radial electric field in high-T_e electron root plasmas on LHD

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The evolution and parameter dependence of a core positive radial electric field is examined in high-T_e electron cyclotron heated (ECH) discharges on the Large Helical Device (LHD). These low collisionality plasmas develop a region of reduced transport in the core leading to high electron temperatures (T_{eo} = 5–10 keV) at moderately low densities (n_{eo} = 1–2 x 10^{19} m^{-3}). Changes in the injected ECH power and electron density have a strong effect on the structure of the radial electric field. The installation of the X-ray imaging crystal spectrometer diagnostic (XICS) has allowed measurements of the core plasma poloidal rotation, which can be directly related to the radial electric field, without the need for neutral beam injection. These measurements allow the characterization of the core radial electric field (E_r) at a range of injected powers and plasma densities. The time evolution of E_r during changes to the input power is also investigated, showing an expansion of the core radial electric field as the ECH power is increased and density decreased. The E_r measurements from the XICS system are compared to neoclassical predictions from GSRAKE and FORTEC-3D, which predict the observed strong positive (electron-root) E_r. The absence of neutral beam injection in this study simplifies neoclassical analysis as there is no external momentum input or fast ion population, as is assumed in the calculations. Power scans in this study were done using fast ECH modulation, a novel technique for investigations on LHD. Finally the implications of these results on ion and electron heat transport is investigated using the TASK3D transport suite.

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Shear Alfvén continua and discrete modes in the presence of a magnetic island

S1: Impacts of magnetic topology / 3D effects

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In this work, we investigate the effects of a magnetic island on the properties of the Alfvén spectrum. In order to accomplish this task, novel use of 3D MHD equilibrium tools and analytic approaches are employed. This work provides an explanation for previously unexplained Alfvénic activity present on beam-heated Madison Symmetric Torus (MST) experiments.

Analytic techniques can be used to describe the effect of a magnetic island on the shear Alfvén continuum [1]. Using an island coordinate system and a WKB approximation of the linearized ideal MHD equations, the island is shown to cause an upshift in the accumulation point frequency. The minimum of the frequency spectrum is shifted from the rational surface to the island separatrix. This analytic result confirms previous numerical work on this topic [2].

The theory is used to explain some previously unresolved Alfvénic activity observed on MST during neutral beam injection [3]. A sizable n=5 island exists in the plasma core that has not been included in past simulations of the Alfvén spectrum in MST. The theoretical Alfvén continua in the core of the island provide a gap in which the observed n=4 Alfvénic bursts reside, suggesting that these modes may arise from a coupling due to the island. Numerical simulations using the STELLGAP/AE3D codes [4], as well as a new code called SIESTAlfvén have identified the bursts as the first observation of an Island-induced Alfvén Eigenmode (IAE). The IAE arises from a helical coupling of mode numbers, similar to the helicity-induced Alfvén eigenmode, but occurs in the core of an island. The SIESTAlfvén results come from 3D MHD equilibria with an island for MST obtained using the SIESTA code [5]. The Alfvén modes are computed by solving the generalized eigenvalue problem obtained from the Hessian matrix of the potential energy along with the inertia matrix of the SIESTA equilibrium.

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Radial Correlation Length Studies by Doppler reflectometry in TJ-II stellarator: experiments and simulations

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Experimental and computational studies have been performed to characterize the radial correlation length (Lr) profile of the plasma turbulence under the presence of magnetic islands. It is well known that the interpretation of correlation measurements by reflectometry is challenging in terms of real turbulence radial correlation length. For that purpose, theoretical [1], computational [2] and experimental [3] studies were carried out to find a way to determine properly Lr using Doppler reflectometry. These studies conclude that under certain conditions it is possible determine Lr through Doppler reflectometry.

One of the aims of this work is to know the behavior of Lr along the plasma radius under the presence of magnetic islands. To achieve this, plasma simulations have been carried out by resistive MHD code; where the magnetic geometry is modeled as cylindrical and magnetic islands like g-modes instabilities [4][5]. In addition, a Full-Wave code [6] acting as Doppler reflectometer synthetic diagnostic was adapted to include the MHD code generated turbulence in order to compare the radial correlation results with those obtained using the TJ-II Doppler reflectometer [7]. In the simulations the Lr profile exhibits a characteristic shape, which is described hereinafter: Lr shows a non-monotonic behavior along the plasma radius with maxima and minima associated to the presence of different magnetic islands. At the island boundaries the correlation length measured towards the island interior is longer than that of the background turbulence, a signature that would allow the identification and characterization of the magnetic islands in the experiments.

In order to confirm experimentally this behavior a set of experiments has been programed in TJ-II using Doppler reflectometer. This system has two independent settable frequency channels required to carry out the radial correlation studies. In particular for this study, the first channel is working at fixed probing frequency while the second one is working in frequency hopping mode at neighboring frequencies. The experiments are programmed to explore several magnetic configurations in order to characterize the Lr profile behaviour at different low-order rationals. The experiments are currently ongoing.

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TEM turbulence optimisation in stellarator experiments

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With the advent of neoclassically optimised stellarators like Wendelstein 7-X (W7-X), turbulent transport becomes relatively more important, and is expected to be dominant in the outer parts of the plasma. Linear analytical theory [1] suggests trapped-electron modes (TEM) are stable in large parts of parameter space when all trapped particles experience bounce-averaged "good curvature", which is the case in perfectly quasi-isodynamic stellarators with the maximum-J property. It has been found numerically [2] that even configurations like W7-X, where a small but finite fraction of the trapped orbits experience bad average curvature, benefit from enhanced TEM stability. Two important questions then arise: Does the enhanced linear stability result in reduced transport? And can this knowledge be used to optimise stellarators not only for neoclassical, but also for turbulent transport? Both questions are addressed here: We present first-of-a-kind density-gradient-driven TEM turbulence simulations demonstrating that this enhanced stability does indeed result in reduced transport compared with a typical tokamak (ITG transport is however comparable unless there is a finite density gradient present). We also present an optimisation method for configurations like the TEM-dominated stellarator HSX, where the growth rates of TEMs are significantly higher than in W7-X. A "proxy" function was designed to estimate the TEM growth rate, allowing optimal configurations for TEM stability to be determined with the STELLOPT code [3] without extensive turbulence simulations. As a proof-of-principle, we show an optimised equilibrium that was evolved from the initial HSX. This optimised equilibrium has reduced TEM growth rates over a broad range of wave numbers and density gradients, and, more importantly, significantly reduced heat flux levels. Since this proof-of-principle configuration is not accessible with HSX's given coil set we attempt an optimisation with STELLOPT where only the magnetic-field coil currents are adjusted. This way the optimisation routine also provides means of exploring TEM turbulence in the accessible configuration space of current stellarator experiments.

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Impact of isotope species and collisionality on ITG and TEM instabilities in helical plasmas

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Understanding the impact of hydrogen-isotope species on micro-instabilities and turbulent transport levels is an important issue in both tokamak and helical/stellarator plasmas. Improved thermal confinement for the deuterium plasma compared to the hydrogen one has been observed in tokamak H-mode experiments[1], and such an isotope effect is examined in the next deuterium plasma experiments in Large Helical Device(LHD).

In this study, micro-instabilities in an LHD-like helical plasma are investigated by using a multi-species electromagnetic gyrokinetic code GKV, where an improved multi-species collision operator is implemented[2]. The transition from TEM to ITG modes is identified by comprehensive parameter scans for $T_{\rm e}/T_{\rm i}$, $R_{\rm ax}/L_{T_{\rm e}}$, $R_{\rm ax}/L_{T_{\rm i}}$, $R_{\rm ax}/L_n$, and the magnetic configuration. Also, the effects of isotope species with different mass number $A_{\rm s}$(=1 for Hydrogen:s=H, =2 for Deuterium:s=D, =3 for Tritium:s=T) on the ITG and TEM instabilities, including its collisionality dependence, are examined. It is clarified that significant dependence of the isotope species appears in the reduction of the normalized growth rate $\overline{\gamma}_{\rm s} \equiv \gamma_{\rm s} R_{\rm ax}/v_{\rm ts}$ for TEM [Fig.1(a)], while the ITG growth rate is almost independent of the isotope species [Fig.1(b)]. Such strong dependence in TEM is resulting from the ion-mass dependence of the ratio of the electron-ion collision frequency to the ion transit frequency, which is essential for the collisional stabilization of TEM. The collisionality(ν^*) scan also reveals that, only for TEM cases, there exists a ν^*-regime ($\nu^*_{\rm ei} \gtrsim 0.04$ in the present case) leading to the reduction of the mixing-length diffusivity $\gamma/k_\perp^2$ [Fig.1(c)]. Dependence of the collision operator model and evaluations of the quasilinear heat and particle fluxes are also presented.

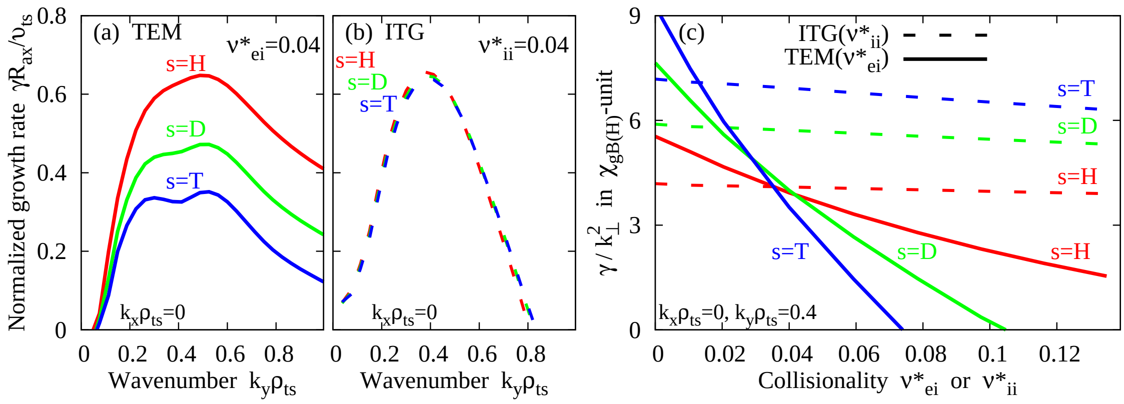

Fig.1: Normalized growth rate $\gamma R_{ax}/v_{ts}$ ($\equiv \overline{\gamma}_s$) for (a)TEM and (b)ITG (s: isotope species). (c)ν^*-dependence on $(\gamma/k_\perp^2)/\chi_{gB(H)} = \sqrt{A_s}\,\overline{\gamma}_s/\overline{k}_{\perp s}^2$, where $\chi_{gB(H)} \equiv \rho_{tH}^2 v_{tH}/R_{ax}$, $\overline{k}_{\perp s} \equiv k_\perp \rho_{ts}$.

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The characteristics of long range correlation in Heliotron J

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Zonal flow is considered to play a crucial role on the formation process of transport barriers and can determine the core plasma performance. Zonal flows appear as a radial electric field fluctuation with axisymmetric structure in toroidal/poloidal directions in the low frequency range of ~ kHz, and has finite wavelength in radial direction with typical wavelength of 1 cm, which is 10-50 times larger than the ion Larmor radius [1, 2]. The spatio-temporal structures of the zonal flows are basically similar in many devices, however, slight differences on the characteristics are also reported on a few papers, for example, regarding strong correlation in the broad frequency range up to 20 kHz or longer radial wavelength [3, 4]. Further comparative studies focusing on such differences for the behaviour of zonal flows are required, which gives implications for the development of future fusion machines with better confinement properties from the aspects of zonal flow – turbulent system.

In Heliotron J, a toroidally symmetric radial electric field (E_r) fluctuation has been observed using Langmuir probes, which has quite similar characteristics to zonal flows except for the longer radial wavelength [5]. In spite of the long radial wavelength, the fluctuation can produce oscillating velocity shear because the correlation length of the symmetric fluctuation in radial direction is about ~ 1-2 cm, which is favourable to generate the velocity shear and is also consistent with the observation in other devices. In the result of cross-bicoherence analysis, significant value was observed in the low frequency range corresponding to the Er fluctuation. Cross-correlation analysis shows that radial and poloidal electric field fluctuations in the broadband frequency range show clear differences on the property at the boundary of LCFS, which contribute to the steep increase of Reynolds stress (R.S.). In addition, the profile of R.S. has a similar structure as the fluctuation amplitude of the symmetric *Er* fluctuations. A statistical analysis using two dimensional probability density function (2D-PDF) was used to investigate the responses of the symmetric fluctuation and R.S., and the PDF shows different profile depending on the positive/negative value of δEr induced by the symmetric fluctuation, which suggests the symmetric *Er* fluctuation and R.S. are coupled each other. These observations are well consistent with the scenario of zonal flow generation through nonlinear interactions. In this presentation, we will discuss the detailed characteristics of the long range correlation observed in Heliotron J, and the differences from those in other devices.

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Simulation Study of the Deuterium Experiment Plasma of LHD by TASK3D and GNET

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The deuterium experiment project from 2017 is planned in LHD, where the deuterium NBI heating beams with the power more than 30MW is injected to the deuterium plasma. Main objects of this project are to make clear the isotope effect on the heat and particle transport in the helical plasma, and to study energetic particle confinement in a helical magnetic configuration measuring triton burn-up neutrons.

We have developed an integrated transport simulation code, TASK3D[1], and applied to the high ion temperature plasma of LHD[2, 3]. The particle and heat transport equations are solved and compared with LHD experimental results. Heat and particle transports are assumed to be the sum of the neoclassical transport and the turbulent transport. Relatively good agreements are obtained between the simulated and experimental radial profiles of the density and temperature in the steady state plasma of LHD[2, 3]. Also, in order to study the kinetic behavior of the energetic particle in a helical magnetic configuration, we have developed GNET[4] code, in which the drift kinetic equation of energetic particles is solved in five-dimensional phase space. GNET code is improved to evaluate the D-D and D-T fusion reaction between the energetic and thermal ions.

In this paper we study the deuterium experiment plasma of LHD by TASK3D and GNET. We solve the heat and particle transport of the deuterium plasma assuming the turbulent transport improvement models by the isotope effect; a simple and the H-He experiment based ones, from the turbulent transport model for the hydrogen plasma. The radial profile of density, and electron and ion temperatures of the deuterium experiment plasma are evaluated. Next we study the production and confinement of tritons (1.01MeV) by GNET code. Characteristic distributions of tritons in the real and velocity spaces are shown. We also evaluate D-T fusion reaction rates and triton burn-up neutron signals of the diagnostic system using the obtained triton distributions.

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Progress in applying gyrokinetic heat diffusivity model to transport simulations for helical plasmas

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A reduced transport model for the turbulent ion heat diffusivity due to the ion temperature gradient (ITG) mode was obtained from the gyrokinetic simulation using the GKV-X code for the high-T_i Large Helical Device (LHD) plasmas [1]. This model is given by the function of the normalized zonal flow decay time $\tilde{\tau}_{ZF}$ and the mixing length estimate $\tilde{\gamma}_{\tilde{k}_y}/\tilde{k}_y^2$ integrated over the poloidal wavenumber space, where $\tilde{\gamma}_{\tilde{k}_y}$ is the normalized linear growth rate of the ITG mode. The nonlinear gyrokinetic transport simulation results are quantitatively reproduced by the reduced model, which we call the gyrokinetic heat diffusivity model. However, it is costly to carry out linear calculations of the growth rate by the gyrokinetic simulation at each time step of the transport simulation. How to apply the turbulent heat diffusivity model derived from the gyrokinetic simulations to the dynamic transport simulation with a low computational cost was shown [2]. This additional modeling for the turbulent ion heat diffusivity has been applied to the integrated code, TASK3D.

The TASK3D simulation of the high T_i plasmas with the carbon pellet injection in LHD was performed [3] using the gyro-Bohm+∇T_i model for the turbulent ion transport. In this study, the method to apply the reduced model to the transport simulation is used for the LHD shots different from that in [2]. Dependence of the density scale length on the turbulent heat diffusivity is examined in a case of the pure hydrogen plasma. We examine the ion temperature profiles before and after the C pellet injection for applying the gyrokinetic heat diffusivity model to TASK3D. The neoclassical transport is numerically estimated for the multi-ion species plasma in the TASK3D simulation. Introduction of the carbon impurity density remains to be improved in the turbulent transport studies of the LHD high T_i plasmas. The simulation results without the effects of the carbon density on the turbulent heat diffusivity are compared with the experimental results. The effect of the multi-ion species plasma on the heat diffusivity after the C pellet injection is investigated.

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Profile sensitivity of turbulent transport in LHD plasmas

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The transport of heat and particle is mainly caused by micro-scale turbulence due to the temperature and density gradients in magnetized plasmas including tokamaks and helical systems. In the Large Helical Device (LHD), the local gyrokinetic turbulence simulations with adiabatic electrons were performed by GKV-X code [1]. The simulations reproduce the ion heat flux which is comparable to the experimental results, except for the outer radial region where the simulation results show the smaller heat flux than the experimental results. The under-prediction is similar to the L-mode tokamak case which is known as so-called *shortfall* issue. Recently, the gyrokinetic simulations within the experimental errors of the plasma profiles have been performed to remove the ion-scale shortfall issues because of the profile stiffness to the gradients of temperature and density [2].

In this work, we perform the gyrokinetic analyses for the ITG and TEM turbulent transport with adiabatic electrons or kinetic electrons in the LHD plasmas. The sensitivities of the transport fluxes to the plasma profiles are discussed within the experimental error ranges of the temperature and density profiles. Figure 1 shows the ion heat diffusivities obtained by the ITG simulations with the adiabatic electrons and the experiment. In outer radial region, the simulations within the experimental errors of the ion temperature (T_i) can cover the experimental diffusivity, because the experimental T_i errors enhance the range of the simulation results. On the other hand, in the core region, the simulation results have larger ambiguity rather than the outer region, due to the fact that the experimental T_i errors are larger in the core region.

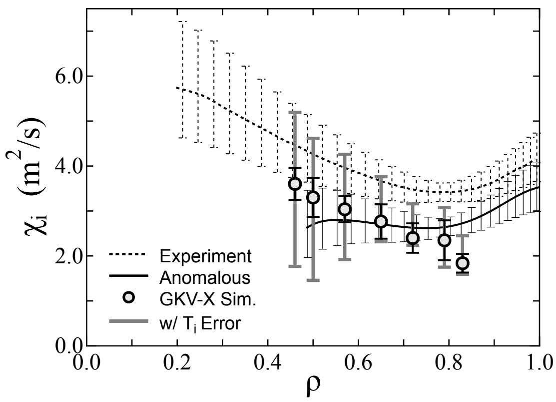

Figure 1: *Ion heat diffusivity obtained by gyrokinetic simulations (symbols) and the experiment (solid curve). The simulations within the errorbars of temperature profile (gray error bars) enhance the range of the diffusivity.*

In the outer radial region, there exists large density gradients which may cause the TEM instability, while the density profile in core region is almost flat. Therefore, we perform the gyrokinetic simulation for the TEM turbulence with kinetic electrons in the outer radial region within the errorbars of the experimental density profile with the precise collision operator [3].

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Gyrokinetic analysis of turbulent particle and heat transport in low- and high-Ti phases in a discharge of LHD

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Validation of gyrokinetic simulations for turbulent transport is one of the critical issues in predicting the performance of stellarator-heliotron systems. Turbulent transport in a high-ion-temperature discharge of the Large Helical Devices (LHD) was investigated by means of gyrokinetic simulations with the adiabatic electron model [1] and with the full kinetic electron model [2]. It is demonstrated that the temperature gradient length reproducing the heat flux observed in the experiment is predicted within 20% error to the experimentally observed value, and turbulent particle flux has a pinch effect [2].

Some issues are still unresolved in the previous gyrokinetic analysis with the adiabatic electron model. One is that the experimentally observed finite turbulence transport at a low temperature phase (t=1.8s) [3] is not reproduced by the simulation because the ion temperature gradient (ITG) mode is stable or weak.

In this work, it is shown that the kinetic electron effects enhance the growth rate of the ITG mode and lead to a finite turbulent transport in a low temperature phase (t=1.8s). Figure 1 shows the linear growth rates of instabilities in a low-Ti phase (t=1.8s, Ti0=1.7keV) and in a high-Ti phase (t=2.2s, Ti0=3.9keV) in the LHD discharge (the shot number 88343). It is found that the ITG modes with the kinetic electron model (KE) are more unstable than those with the adiabatic electron model (AE). Nonlinear simulations show that the energy fluxes in the low-Ti phase (t=1.8) are finite at rho=0.68 and are in good agreement with the experimental observation (Fig. 2). The ion energy flux at t=2.2 is close to the experimental observation when the temperature gradient is reduced 20% in the simulation.

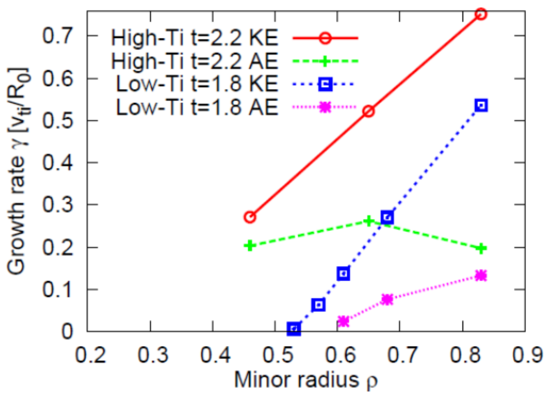

Fig. 1 Linear growth rates.

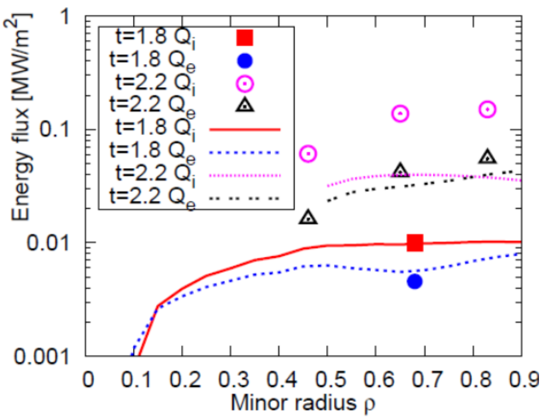

Fig. 2 Ion and electron energy fluxes due to turbulence, Qi and Qe. Symbols are from the simulations, and curves are from the experiments.

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Effects of Radial Electric Fields on Linear ITG Instabilities in W7-X and LHD

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The global (full radius, full flux surface) gyro-kinetic particle-in-cell (PIC) code EUTERPE is used as an efficient tool for numerical simulation of plasmas in three-dimensional (3D) magnetic fields. EUTERPE is capable of simulating up to three kinetic species - ions, electrons and fast particles/impurities. The code solves the field equations for the electrostatic and parallel vector potentials and can be used for linear and nonlinear simulations. EUTERPE is routinely applied to 3D equilibria obtained with the VMEC equilibrium code.

Radial electric fields are considered to be important for the suppression of ion-temperature-gradient (ITG) driven modes in fusion devices. Most stellarators exhibit no or slow toroidal rotation, and the occurrence of radial electric fields is connected with a background pressure gradient. To supplement the earlier characterization of linear ITG behaviour, this contribution will present results for the impact of radial electric fields on growth rates, localization and structure of ITG modes to be expected in W7-X and LHD, in particular highlighting the global nature of these effects.

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Magnetic null point modelling in BOUT++

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Simulations of instabilities and turbulence in X-point configurations are challenging due to the limitations of field-aligned coordinate systems: X-point dynamics are often extrapolated based on nearby flux surfaces, which could exclude relevant physics. Here we present the results of 3-dimensional driftwave turbulence simulations in X-point configurations using coordinate systems which are not aligned to the magnetic field. The studies have been performed using a cold ion fluid model originally developed for blob studies [1,2] which evolves plasma density, vorticity, parallel velocity and parallel current density. The effects of an external X-point field have been explored in simulations originally intended to determine the feasibility of experimentally studying X-point configurations in a linear plasma device [3]. These studies have been extended to toroidal geometries by simulating blob dynamics in complex configurations within the TORPEX device [4], as shown in the figure below. Additionally, the Flux Coordinate Independent (FCI) [5] approach to calculating parallel derivatives has been implemented into BOUT++ . The main advantage of this scheme is that it does not rely on the structure of the magnetic field; provided the field is defined at a given grid point, the parallel derivatives can be calculated there. This allows for complex magnetic geometries including X-points. The extension of these methods to nonaxisymmetric geometries and their applications to stellarators and heliotrons will be discussed.

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Preliminary Findif calculations for limiter W-7X configuration

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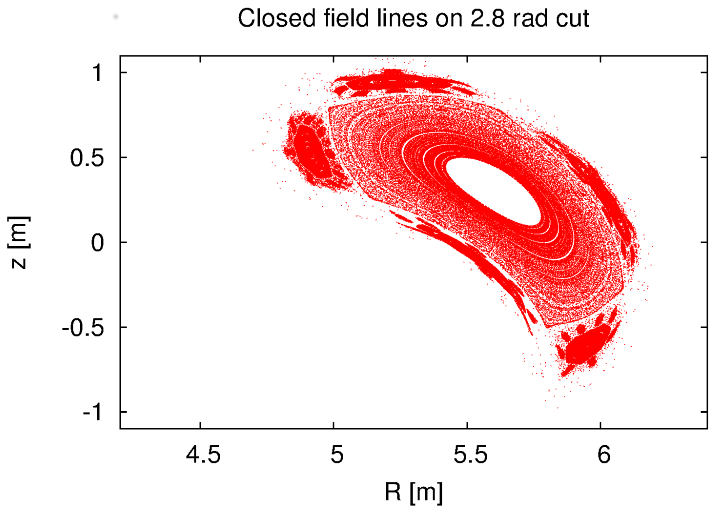
Findif is a 3-D finite difference, multi-fluid code devised (mainly) for edge plasma simulations. Intrinsically and irreducibly 3-dimensional problems, such as stellarator or ergodic divertor [1] simulations, are natural play ground for Findif.

The mesh for Findif is produced in $\vec{B}$ field line tracing process and thus it directly takes into account magnetic field structure. Lines in the core area, magnetic islands, ergodic region and region of open field lines need to be carefully chosen so that they cover space quite uniformly. Computational mesh nodes lie where field lines cross polodal cuts or material surfaces, but on the latter ones boundary conditions are imposed. While parallel transport happens along field lines, perpendicular one is present on the Poincaré plots. Irregular mesh there precludes using simple derivatives computation methods; nonconservative "free point method" is currently used instead.



Closed magnetic surfaces found during tracing procedure. A Poincaré plot.

Strong parallel transport is naturally separated in the equations solved, which eliminates an important source of numerical errors. Simplified (fluid) treatise of neutrals dynamics allows for fast runs. The code presented is currently being applied to limiter configuration of Wendelstein device and first results of consecutive code modules for several scenarios are presented.

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Doppler Reflectometry for the first plasmas of W7-X

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A V-band hopping reflectometer [1] is being installed at W7-X for Doppler reflectometry (DR) measurements at the plasma periphery. The V-band DR system aims for providing plasma turbulence and flows measurements already at the first W7-X plasmas, contributing to the confinement and transport studies carried out for the OP1.1 plasma scenarios. To that end, a versatile Gauss antenna is used with circular horn output that allows both x- and o-mode with an extremely broadband design covering V-band and W-band, 50-110 GHz [2]. A stand-alone fast DAQ system consisting in an industrial PC with a PCI data acquisition board (PCI-DAS4020) is used together with the V-band hopping reflectometer. The DAQ system has been designed to account for long data records at high sampling frequencies (up to 18 s @ 20 MHz, 2 channels) required for Doppler reflectometry measurements. The timing sequences required to operate the reflectometer and the DAQ systems are generated by a second PCI board (PCI-1780U) incorporated into the PC. This board is a programmable 8-ch, 16-bit Counter/Timer Universal PCI Card. Data storage within the W7-X database is carried out by a MDSplus-based software. All drivers required for the operation of the V-band reflectometer: the Counter/Timer PCI and the PCI-DAS boards have been also developed. Full control of the reflectometer is performed via a Graphical User Interface (GUI) which has been specially developed for this purpose. Three different functionalities are integrated into a single user interface, a) to setup the reflectometer working frequencies, ii) to define the DAQ parameters and iii) to handle the timing sequence required to operate the reflectometer. Data visualization and analysis is performed with the help of dedicated software developed in MATLAB®. This data analysis UI also offers an interface to the TRAVIS ray tracing code [3] allowing the radial localization of DR measurements and the determination of the wave-numbers probed by the reflectometer. The installation of a second V-band hopping reflectometer for Doppler reflectometry (DR) measurements toroidally separated from the first one is planned after OP1.1 for zonal flow measurements. The physics conceptual design for the second Doppler reflectometer has been carried out using the ray tracing code TRAVIS [3]. The applied design optimization procedure is based on the minimization of the parallel wavenumber contribution to the measurement and ensures good performance for a set of standard magnetic configurations in W7-X.

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Measurements of plasma density in Uragan-2M torsatron using dual-polarization interferometry

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Ordinary wave interferometry is a well known and commonly used plasma diagnostics for fusion devices and plasma technology [1, 2]. The ordinary wave number depends only on plasma density in the case of perpendicular probing with respect to the main magnetic field. Thus, the ordinary wave phase shift is proportional to the line-integrated plasma density along the chord of probing for the wave frequency greater than plasma frequency. For the extraordinary wave perpendicular probing, the phase shift depends on plasma density and the confining magnetic field distributions. Since the magnetic field is known for stellarator/torsatron devices, additional information about plasma density profile may be inferred from the extraordinary wave phase shift measurements. For example, this may be the peakedness of the plasma density profile $\bar{n}/n_0$, where n_0 is the plasma density on magnetic axis and $\bar{n}$ - averaged plasma density.

The dual polarization interferometer has been designed and installed in Uragan-2M at the cross-section where magnetic surfaces are vertically elongated. For data analysis, the set of phase shifts is calculated using real spatial dependence of the magnetic field and the following parameterization of plasma density, $n(\psi) = n_0\left(e^{\xi} - e^{\xi\psi}\right)/\left(e^{\xi} - 1\right)$. Here ψ is the flux surface label equal to 0 at the magnetic axis equal to 1 at the last closed magnetic surface, ξ is profile peaking parameter. Calculated phase shifts are matched with experimental results in order to determine the central density and peaking parameter of the Uragan-2M plasma. Thus, rather simple and effective way of plasma density measurement has been implemented.

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Conceptual Design of a Heavy Ion Beam Probe for W7-X

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An international scientific partnership aimed at studying the characteristics and evolution of electrostatic fluctuations and electric field, as well as their role in transport and flows, has been unfolding between IPP and Xantho Technologies. Implementation of a heavy ion beam probe (HIBP) diagnostic on Wendelstein 7-X (W7-X) is central to this endeavor.

Advancing an understanding of particle and energy transport in the 3D configuration of optimized stellarators requires direct measurements of amplitudes, wavelengths, cross-phases, correlation lengths, and characteristic time scales, in concert with theory and computation. The HIBP is uniquely well suited for this task; it is able to acquire temporally and spatially resolved measurements of $\tilde{n}/n(r)$, $\tilde{\varphi}(r)$, wavenumbers ($k_{nn}, k_{\varphi\varphi}$), cross coherence and phase ($\gamma_{n\varphi}$, $\alpha_{n\varphi}$), electric potential $\varphi(r)$, and radial electric field $\boldsymbol{E}_r(r)$. The HIBP is capable of acquiring these quantities simultaneously in the plasma interior [1].

We have initiated the work necessary to implement an HIBP on W7-X during the second diagnostic phase (OP2). This includes two fundamental elements of diagnostic development - *hardware interface characterization* and *foundational beam simulations*. Hardware interface characterization includes consideration of - port pairs for the injection and detection of the ion beam, location of the accelerator and analyzer, and flux of particles and radiation from the plasma. Foundational ion beam simulations will consider the beam trajectory as a function of space and time to and within the plasma. These simulations use magnetic and modeled electric field data as well as estimates of expected temperature and density within the plasma. These will merge within trade-off analyses to identify the required accelerator energy, suitable beam ions, energy analyzer requirements, and cross-section of the plasma accessible to measurement. These fundamental elements of diagnostic development will enable us to formulate the conceptual design of an HIBP for W7-X.

This endeavor will use components of the 2 MeV HIBP diagnostic which was installed previously on the TEXT-U tokamak and is now in Greifswald. The system, which was developed through a US-DoE grant, was transferred to IPP for use on W7-X following the shutdown of TEXT-U. Xantho brings U.S. HIBP expertise and detailed knowledge of this hardware to this collaboration.

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Design of a phase contrast imaging diagnostic for Wendelstein 7-X

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A phase contrast imaging (PCI) diagnostic for Wendelstein 7-X is being designed and constructed by collaborators at the Massachusetts Institute of Technology, a project funded by the US Department of Energy. We present an overview of the PCI diagnostic design with an outlook to future extensions of the diagnostic that will include localization capability. The PCI technique is an interferometer with an "internal" reference beam that measures fluctuations in the electron density, integrated along the beam path. Information regarding the fluctuations is carried in the scattered components of the beam, with the scattering angle is proportional to the wavenumber of the fluctuations orthogonal to the direction of propagation of the beam. The PCI diagnostics installed at the DIII-D and Alcator C-Mod tokamaks have lower and 3dB frequencies of 5 kHz to 600 kHz and can detect fluctuations with wavenumbers in the range of 1 cm^{-1} to 20 cm^{-1}, approximately. The PCI diagnostic will be one of few fluctuation diagnostics in the early phases of W7-X operations, and therefore positioned to provide valuable insights into the relation between confinement and turbulence. We plan to have a PCI system installed and operational by the start of the OP-1.2 campaign, with further developments to continue afterward.

We will present a design and analysis of the PCI system given the assessment of and the available space within the W7-X torus hall and the overall measurement objectives. The complexity and sensitivity of the optics will largely be determined by whether the detector and optics are placed in a remote diagnostics lab or located proximate to the machine in the torus hall. The structure of turbulent fluctuations as measured by the PCI diagnostic will be modelled using a synthetic diagnostic analysis of simulations of ITG turbulence and will be used to assess the ability to localize electrostatic fluctuations along the beam path based on the variation of magnetic pitch angle. Preliminary analysis of the magnetic field geometry, computed with the code VMEC by our collaborators at PPPL, indicates that the total magnetic pitch angle variation along the PCI beam path will be roughly 20 degrees and should be sufficient for separation of core and edge signals.

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Study of turbulence rotation and local ι using Correlation Reflectometry at W7-X

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The energy and particle transport in the edge of magnetic fusion devices is dominated by the spatiotemporal dynamics of turbulence. A better understanding can only be gained, if sophisticated experiments are compared to numerical simulation results. With large compoutational effort the spatial structure of turbulence can be computed on small time scales using gyrokinetic simulations. But today diagnostics still suffer from temporal and spatial resolution that allow for a detailed comparison.
The scope of this presentation is to outline a diagnostic for the measurement of the (i) poloidal velocity of turbulent structures by using Poloidal Correlation Reflectometry (PCR) and (ii) measuring the poloidal (perpendicular) structure size. The diagnostic has been recently installed at W7-X. It consists of a set of 5 antennae which are located slightly below the outer midplane in a cross-section where the plasma has a bean like shape. The wave guides are connected to a heterodyne reflectometer operating in *Ka*-band and in O-mode polarization. One launch and four receiving antennae, poloidally and toroidally separated, allow not only to measure the propagation of turbulent structures at four different positions but, also the measurement of the the magnetic field line inclination. This method was applied for the first time at TEXTOR [1] and is recently used at ASDEX Upgrade to estimate the local safety factor q. Besides turbulence rotation the measurement of turbulence properties as decorrelation time and poloidal correlation length are within the scope of the instrument.
For an overview of turbulence phenomena in the plasma of W7-X the instrument will measure turbulence spectra in the first operation phase of W7-X. An interesting topic will be the search and the study of quasi coherent modes, which are widely observed in tokamaks. The origin of this modes is not yet clear and the observation in current free plasmas could help for a better understanding of these modes.

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Fluctuation measurements through correlation radiometry and reflectometry on Heliotron J

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Non-invasive diagnostics that are capable of simultaneously resolving multiple spatial and temporal scales are necessary to study the interaction between turbulence and transport. The recent development of high speed oscilloscopes with bandwidths up to 100 GHz [1] has enabled the direct measurement of microwaves during plasma experiments, introducing a new method to measure electron temperature and plasma density fluctuations at multiple radial scales [2]. A Teledyne LeCroy oscilloscope with 36 GHz bandwidth (80 GS/s) has been used as part of a heterodyne radiometer, and separately as part of a reflectometer system, to make fluctuation measurements on Heliotron J. The oscilloscope was used to directly measure the intermediate frequency spectrum from the radiometer, corresponding to 2nd harmonic Electron Cyclotron Emission (ECE) at 58-74 GHz, and digital correlation techniques were applied to determine the correlation length of electron temperature fluctuations near the core of the plasma. A conventional single-sightline correlation ECE (cECE) radiometer [3] has been developed and implemented on Heliotron J, and results from the digital and conventional cECE will be compared. During two-source microwave reflectometry experiments, the oscilloscope was used to directly measure the launched and reflected waves between 24-28 GHz, corresponding to the steep plasma density gradient region of Heliotron J, and the radial correlation length of the plasma density fluctuations will be presented.

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Plans to Use Thomson Scattering to Resolve Centimeter-scale Fluctuations and Electron Pressure Gradients near the Last Closed Flux Surface in the W7-X Stellarator

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Transport near the last closed flux surface (LCFS) of the stellarator is thought to be governed by drift-wave turbulence arising from ion temperature gradient (ITG) and/or trapped electron modes (TEM). In some cases, transport barriers are observed that result in locally steep pressure gradients in this region. In addition, gyrokinetic codes predict poloidal (and toroidal) localization of ITG turbulence on the scale of centimeters in the outer regions of the W7-X Stellarator. We present plans to measure the electron pressure gradient, and possibly fluctuations, using Thomson scattering in the region of the LCFS in W7-X. Thomson scattering provides temporally ($\Delta t \sim 10$ ns) and spatially ($\Delta r_{eff} \sim 1$ cm) localized measurements of electron density and temperature. The W7-X Thomson system could be configured with as many as five (Nd-YAG) lasers in a diverging fan to resolve spatial variations in the poloidal plane. The scattering volume elements imaged by the collection optics from such a fan near the LCFS can provide detailed equilibrium profile measurements (relative to flux surfaces) and perhaps spatial scale-length (in the poloidal direction) measurements of fluctuations. Time-delayed laser pulses along a fixed beam path may also provide information on the frequency spectrum of the fluctuations. Predicted Thomson scattering signal levels in W7-X suggest that fluctuation amplitudes in electron density and temperature larger than 5% may be measurable.

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Review of biennial stellarator activity in A.M. Prokhorov General Physics Institute

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Review of theoretical and experimental works on stellarator physics that were performed over the past two years is presented. Experimental works were carried out on large shear, medium size L-2M stellarator with gyrotron complex MIG-3 that was completely set in operation in 2013.

The phase shift between fluctuations of the electric potential and plasma density in the edge turbulence was analyzed. It is shown that the observed phase shift Ω varies in a wide range from π to 0, gradually decreasing with deepening inside the plasma. We note that, within the framework of the magnetohydrodynamic theory, plasma was definitely unstable with respect to resistive interchange modes in all cases under study. It is demonstrated experimentally that the widespread notion that the phase shift $\Omega \approx \pi/2$ is characteristic of only resistive interchange modes is hardly universal. The experimental results are analyzed on the basis of analytical estimates and calculations.

Non monotonic density profiles were observed. This effect has definite threshold in heating power. It is shown that pressure profile at this, within measurement accuracy, remains monotonic function of magnetic surfaces average radius.

It is shown that axisymmetric oscillations (ASO) with finite poloidal mode number were observed at specific conditions, namely, when the sufficiently large (in absolute value) negative current is induced in electron cyclotron heated (ECH) plasmas. In all the cases considered nothing but $m = 2$, $n = 0$ was found. Here, m, n are the poloidal and the toroidal number, respectively. Threshold for ASO appearance can be calculated with reasonable accuracy within classical approach. However, peculiarities of current penetration into plasmas demonstrate that non classical effects are significant. It is shown that at sufficiently large in absolute value of negative current, instability zone (with respect to ideal and resistive interchanges) is formed in the central part of plasma and, hence, can be the reason for the observed degradation of plasma confinement.

Finally, we present analytical calculations of peeling modes. This work was initiated by numerical calculations [1,2], where it was shown that in Mercier unstable plasmas both internal and external peeling modes can be unstable, moreover, growth rate of external mode is an order of magnitude grater than that of internal mode [1]. For Mercier stable plasmas external peeling modes were found [2]. Analytics show that in currentless plasmas internal peeling modes in Mercier stable magnetic hill plasmas are stable, while external peeling modes, can be unstable but under additional condition on them. The calculated thresholds are completely confirmed by experimental results achieved.

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Peeling mode stability/instability condition for Mercier stable magnetic hill configuration

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The plasma edge region is supposed to affect total plasma confinement. Desire to work in high performance confinement regimes requires understanding of edge localized modes that tend to reduce plasma edge pressure gradient. In elegant work by Lortz [1] it was shown that ideal MHD modes can be unstable near plasma boundary of Mercier stable [2] magnetic hill configurations (peeling modes). In [1] magnetic surface was located inside but in vicinity of free plasma boundary. The proving demanded the use of energetic principle, neglect of vacuum contribution and the use of probe function that abruptly drops in a small region between rational magnetic surface and boundary. Shortly after it become clear that Lortz formalism is valid for the external modes (see, e.g. [2]). Nowadays peeling modes are widely explored for interpreting tokamak edge localized modes (peeling ballooning modes) and are a rarity in stellarators, despite the fact that stellarators are basically edge magnetic hill systems. Conventionally, edge localized modes are attributed to resistive interchange activity (see, e.g. [3], a review).

This work was motivated by the following concurrence of circumstances. Numerical calculations were performed [4, 5], where peeling modes were found. Peeling modes were observed in experiments [6]. Lortz formalism [1] also demands scrutiny. For example, probe function abruptly drops on non rational magnetic surface that allows infinite current density there. Condition of peeling mode stability coincides with resistive interchange nonthreshold criterion that is not so good.

We have performed the analysis of the vacuum region role. Currentless plasma is considered. Here, by currentless plasma we mean equilibrium plasma in which the total current flowing through the cross section of each magnetic surface is zero. In this case, the only source of instability is the plasma thermal energy. Plasma pressure is taken to be zero at the plasma vacuum boundary. After the matching of external and internal solutions Newcomb method was applied [7]. As in [1] high poloidal mode number approximation $m>>1$ was used.

It is shown that correct taking into account vacuum region forbids internal peeling modes in Mercier stable magnetic hill plasmas, allows external peeling modes, but impose additional condition on them. This threshold is in reasonable agreement with experimental observations [6] and numerical calculations [5]. The threshold decreases with poloidal mode number m. It is shown however, that higher modes may be stabilized due to finite ion Larmor radius effects. The performed analysis does not forbid internal unstable peeling modes with moderate m. This problem demand separate numerical calculations.

Experimental conditions necessary for peeling mode observation are estimated. Mode excitation, in principle, may serve as a float (as on fishing line) indicating of pressure gradient at the plasma edge over certain value increase. Probably this mode may help in cleaning the discharges from impurities

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Session 3

Interactions among energetic particles, MHD and transport

Challenges Posed by High-Performance Steady-State W7-X Plasmas

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The advanced stellarator Wendelstein 7-X (W7-X) has been designed and equipped to demonstrate the potential of this concept for ultimately providing a steady-state, disruption-free source of fusion energy with only modest demands for recirculating power. Experimental scenarios which enable high-performance plasmas under (quasi-)steady-state conditions will thus have highest priority for the W7-X program once a high-heat-flux divertor has been installed (expected by the end of 2019) to enable exhaust of up to 10 MW of cw ECRH. In this context, high-performance scenarios are those in which reactor-relevant values of all dimensionless plasma parameters (β, ν* and ρ*) are achieved simultaneously. Developing such scenarios will require finding physics solutions to various problems which were not explicitly addressed in the optimization of W7-X; a sample of such problems and possible solutions is the topic of this contribution.

In particular, control of the density profile will require that the particle sources provided to the plasma have profile shapes very similar to those of the energy sources from ECRH deposition. As a consequence, strongly localized heating at the plasma axis will have to be avoided as central refueling (whether by recycling, gas-puff or even pellet gun) is essentially impossible at the densities, in excess of 10^{20} m^{-3} , required for high performance. Pellet refueling poses more general problems as well since ECRH will not be possible during transitory periods in which the local density exceeds the electron cyclotron cut-off. Under certain circumstances, a neoclassical-particle-transport barrier may also appear causing the loss of global density control. These problems are demonstrated by performing simulations using a 1D transport code which also allows the testing of strategies for their avoidance or at least their mitigation. First estimates for an upper limit on the impurity source consistent with steady-state operation of W7-X will also be determined. Avoidance of impurity accumulation is of critical importance for any reactor candidate and is generally considered a particular challenge for stellarators as the negative radial electric field arising in these devices (so-called “ion-root” solution of the ambipolarity constraint) is unfortunately conducive to this accumulation.

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Influence of magnetic well on electromagnetic turbulence in the TJ-II stellarator

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Magnetic well is a key ingredient to stabilise interchange modes in stellarators and, hence, is usually considered as an optimization criterion for those devices, besides neoclassical transport (see e. g. [1] and [2]). The increase of magnetic well is related to the plasma shape, which implies the use of complex coils. Therefore the relaxing of such a criterion can help to design simpler and cheaper stellarators. Magnetic well scan experiments on TJ-II have shown the existence of stable plasmas in Mercier-unstable magnetic configurations [3]. The position of the LCFS is the predicted by the VMEC calculations in all the cases, despite the presence of the n/m=8/5 rational at $\rho \approx 0.8$, the confinement time presents a weak degradation for configurations with magnetic hill, and the electrostatic turbulence shows a moderate increase for those configurations [4]. In this work, we explore the electromagnetic properties of turbulence during the magnetic well scan. Mirnov coils and Langmuir probes are used to detect the modes in the outer part of the plasma ($\rho \geq 0.4$). Three families of modes appear during the experiments: 1) a family of modes of Alfvénic nature with high frequencies (above 100 kHz); 2) a second set of modes of middle frequencies (tens of kHz) and 3) an oscillation at $f \approx$10-20 KHz happens in several cases, mainly in configurations with low magnetic well or magnetic hill, although this behaviour is not systematic. In spite of the fact that the vacuum rotational transform is very similar in all the cases, the Alfvénic mode family changes drastically when decreasing the magnetic well, showing a non-monotonic behaviour of the amplitude, and a decrease of the typical frequencies from 180 kHz in the cases with magnetic well to 60 kHz in the configurations with magnetic hill. This behaviour cannot be explained either considering current or density variations. The differences between the calculated Alfvénic spectra and between the populations of fast particles will be studied as possible causes of such an evolution. Regarding the intermediate frequencies, a coherent mode appears with decreasing frequency as the magnetic well decreases. This mode is assumed to be a GAM, which can survive in these TJ-II plasmas, despite of the strong damping, due to the driving caused by Alfvén modes.

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Computation of continuum damping of Alfvén eigenmodes in fully three-dimensional geometry with multiple-mode coupling

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In ideal MHD, shear Alfvén eigenmodes may experience dissipationless damping due to resonant interaction with the shear Alfvén continuum. This continuum damping can make a significant contribution to the overall growth/decay rate of shear Alfvén eigenmodes, with consequent implications for fast ion transport. One method for calculating continuum damping is to solve the MHD eigenvalue problem over a suitable contour in the complex plane, thereby satisfying the causality condition [1]. Such an approach can be implemented in three-dimensional ideal MHD codes which use the Galerkin method. Analytic functions can be fitted to numerical data for equilibrium quantities in order to determine the value of these quantities along the complex contour. This approach requires less resolution than the established technique of calculating damping as resistivity vanishes and is thus more computationally efficient. The complex contour method has been applied to the three-dimensional finite element ideal MHD code CKA . In this paper we discuss the application of the complex contour technique to calculate the continuum damping of global modes in tokamak, torsatron and W7X stellarator cases. To the authors' knowledge these stellarator calculations represent the first calculation of continuum damping for eigenmodes in a fully three-dimensional equilibrium. The continuum damping of a TAE in a W7X stellarator configuration is found to depend sensitively on coupling to numerous poloidal and toroidal harmonics.

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Plasma Start-up and Wall Conditioning with ECRH in Wendelstein 7-X

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The stellarator Wendelstein W7-X is equipped with a 10 MW 140 GHz ECRH system being the exclusive heating system in the first operation phase OP1.1 starting end of august 2015. The experimental program of OP1.1 will be preceded by a 2 week bake out at 150°C and several days of glow discharges to remove water and impurities from the invessel components. The design of the first wall is a mix of water cooled stainless steel tiles and graphite tiles mounted on a water cooled copper structure. However, OP1.1 is a machine test campaign dedicated to commissioning of the heating system, the diagnostics and the critical components of the device itself, e.g. vacuum vessel and coil system. For this reason, the machine is not fully equipped with wall protecting graphite tiles. Furthermore, instead of the foreseen 10 divertor elements, a graphite limiter is installed on the high field side (HFS) in each bean plane of the five modules preventing a direct contact of the plasma with the unprotected regions of the plasma vessel. To minimize the deposition of sputtered metals on the limiter, helium has to be preferably used in the glow discharges. The associated migration of helium into the lattice of metals and graphite is uncritical for OP1.1, because helium will be also used as discharge gas for the experimental program of OP1.1. The reason is an easier density control. However, W7-X is a superconducting device whose magnetic field is sustained the whole experimental day making intermediate wall conditioning by glow discharges impossible. For this reason, a combination of on- and off-axis ECRH should be used to condition the first wall in the first days of plasma operation and after uncontrolled plasma termination. Six gyrotrons, each connected to a quasioptical launcher, will be available in OP1.1 providing a total absorbed power of about 5 MW in maximum. Even though a discharge should be limited to an overall energy input of 2 MJ because of the uncooled limiter, the high amount of power will guarantee plasma operation even in the case of heavy impurities. Furthermore, extensive plasma start-up studies will be possible in dependence on ECRH power, gas pressure and impurity concentration. The non absorbed power can be analyzed by a stray radiation probe mounted in each of the five modules and an electron cyclotron absorption (ECA) diagnostic, consisting of 128 monomode waveguide antennas incorporated in the HFS graphite tiles opposite to ECRH launchers. The main plasma parameters, electron temperature and density, can be analyzed by an electron cyclotron emission diagnostic (ECE) and a dispersion interferometer which are high priority diagnostics and fully operable from the beginning of OP1.1.

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Rapid NBI plasma initiation using pre-ionization method by non-resonant microwave injection in Heliotron J

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Plasma initiation by NBI alone has been demonstrated in LHD and W7-AS [1,2]. However, in the case that the path-length of NBI is not long enough to initiate the plasma in a limited time, pre-ionization methods are effective in rapid plasma start-up and in reduction in heat load onto the armour tile opposite to the NBI injection. In Heliotron J, medium sized (R/a=1.2/0.17m) heliotron device, the plasma start-up by NBI has been succeeded by the pre-ionization method using low power (<20kW) 2.45GHz microwaves at a magnetic field range of 0.6-1.3 T within the confined plasma volume [3]. In the magnetic field range, there are no fundamental or higher harmonic resonances for the 2.45GHz frequency. The ECE measurement shows the production of the high energy electrons by launching the 2.45GHz microwave. The small amount of the additional gas puffing during the 2.45GHz launch was effective to increase the seed-plasma density. The achieved seed-plasma density was $\bar{n}_e = 4\times10^{18}$ m^{-3} in maximum, being much higher than that of the 2.45GHz O-mode cutoff (7.5×10^{16} m^{-3}). Stochastic acceleration model [4] has been proposed to explain the generation of the high energetic electrons. After the NBI turn-on, an additional gas puff increased $\bar{n}_e$ to over 1×10^{19} m^{-3} in a few tens ms. The successful NBI plasma start-up depends on the seed-plasma density. A clear density threshold ($\bar{n}_e \sim 2\text{-}3\times10^{17}$m^{-3}) has been found to initiate the plasma by NBI. The magnetic field (or configuration characteristics of the high energy particle confinement), the microwave power and the gas pressure are the control parameters for the seed-plasma generation. The experimentally observed density threshold is consistent with a 0-dimensional numerical calculation which is newly developed for the Heliotron J experimental conditions. The pre-ionization method and the obtained seed-plasma density developed in this study have a capability to realize the NBI plasma start-up in experimental devices which have perpendicular NBI such as W7-X [5].

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Limit Cycle Oscillations at the L-H Transition in TJ-II Plasmas

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The spatiotemporal evolution of the interaction between turbulence and flows has been studied close to the L-H transition threshold conditions in the edge of TJ-II plasmas. As in other devices the temporal dynamics of the interaction displays limit cycle oscillations (LCO) with a characteristic predator-prey relationship between flows and turbulence [1]. Recently, some controversial results arise: two types of LCO are found in HL-2A showing opposite temporal ordering [2]. The first type is the standard predator-prey model where the turbulence increase leads the zonal flow generation that suppresses the former. In the second type, the pressure driven ExB flow grows first causing the reduction of the fluctuations. The later points to the pressure gradient as a candidate for maintaining the oscillations and eventually inducing the transition to the H-mode. In the DIIID tokamak, turbulence driven flows trigger the transition to LCO, while the pressure gradient increases later and locks in the H-mode [ref DIII-D]. At TJ-II, the turbulence-flow front is found to propagate radially outwards at the onset of the LCO and in some particular cases, after a short time interval without oscillations, a reversal in the front propagation velocity is observed. Associated to this velocity reversal, a change in the temporal ordering of the LCO is measured. However, the change in the temporal ordering is not related to an intrinsic change in the nature of the LCO. In both cases the turbulence increase leads the process and produces an increase in the ExB flow shear [3]. New experiments are in progress to study the role of the pressure driven flows in the LCO dynamics. The physical mechanisms triggering the onset and radial propagation of the LCO have been also investigated. At TJ-II the LCO are preferentially observed close to the transition threshold conditions at specific magnetic configurations having a low order rational surface located at the inner side of the E × B flow shear location. The behavior of different frequency modes has been analysed and interpreted in terms of a Goedesic Acoustic Mode (GAM) generated by the non-linear mode coupling of Alfvén eigenmodes (AE) that evolves towards a low frequency flow, plus a MHD mode linked to the low order rational surface, as precursors of the LCO

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Global gyrokinetic analysis of energetic particle driven Alfvén instabilities in stellarators

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The GTC global gyrokinetic PIC model has been adapted to 3D VMEC equilibria and provides a new method for the analysis of Alfvénic instabilities in stellarators, 3D tokamaks, and helical RFP states. The gyrokinetic orderings ($k_{\|}/k_{\perp} << 1$, $\omega/\Omega_{ci} << 1$, $\rho_{EP}/L << 1$) are applicable to a wide range of energetic particle driven instabilities that have been observed in 3D configurations. This talk will describe the GTC global gyrokinetic model, its adaptation to stellarators, and recent results. Applications of this model to stellarators have indicated that a variety of different Alfvén instabilities can be excited, depending on the toroidal mode number, fast ion average energy and fast ion density profile. An LHD discharge [1] where bursting $n = 1$ Alfvén activity in the TAE gap was observed has been used as a test case. TAE, EAE and GAE modes have been found in the simulations, depending on the mode family and fast ion profiles used. The dynamical evolution of the instabilities shows the field period coupling between n and $n + N_{fp}$ expected for a stellarator instability (Figure 1). Applications to other devices and the development of gyrofluid reduced models that can capture relevant physics aspects of the gyrokinetic models will also be discussed.

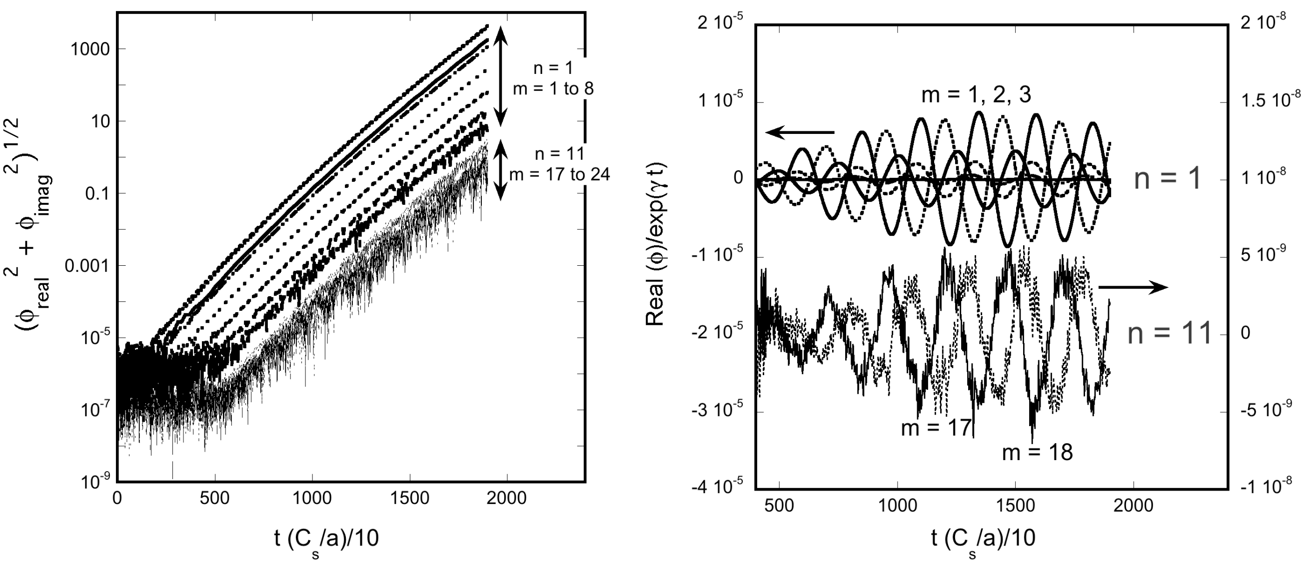

Figure 1 – (a) Amplitude growth, and (b) Alfvénic frequency oscillations × exp(-γt) vs. time, showing mode coupling between n = 1 and n = 11 components, as expected for an N_{fp} = 10 field period stellarator.

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On the effect of localized electron cyclotron heating on NBI-driven Alfvén Eigenmodes in TJ-II

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It has been proven experimentally that localized electron cyclotron heating (ECH) has a clear effect on the NBI-driven Alfvén activity both in the DIII-D tokamak [1] and in the TJ-II heliac-type stellarator [2].

The application of ECH (two 53.2 GHz gyrotrons, up to 250 kW power each) has a complex effect on the character and amplitude of the (otherwise steady) Alfvén Eigenmodes (AEs) measured in TJ-II plasmas. This effect depends on plasma density, ECH power deposition profile, NBI parameters and magnetic configuration. Strong frequency chirping ($\Delta f \approx 20$ kHz) appears for off-axis ECH application (beyond $\rho_{ECH} = 0.34$) with one gyrotron. The chirping amplitude depends on the magnetic configuration, maximum values being apparently linked to a particular value of rotational transform. Further application of additional ECH power with the second gyrotron produces a strong reduction of the mode amplitude.

This work intends to deepen into the physics mechanisms behind these observations. It has to be expected that ECH will affect the drive of the AEs: plasma collisionality is modified through changes in temperature and density profiles and it will affect the slowing down of the fast NBI ions and the dispersion relation of the AE. But it can be also expected that ECH will affect the damping of the AEs because it will alter the trapped electron population whose collisional damping with fast ions would lead to enhanced damping rate [3].

On-going analyses of TJ-II data aim to estimate the importance of the contribution due to plasma profiles. New experiments are being designed, making use of the flexible TJ-II ECH system, in order to scan the trapped electrons population and to measure its possible effect on the collisional damping of the fast NBI ions.

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Low frequency modes associated to magnetic islands in TJ-II stellarator

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Low frequency coherent electromagnetic activity is common in all types of plasmas in the TJ-II stellarator and can be related with the existence of magnetic islands in the confining volume. Two physics processes are characterized: one is island rotation yielding frequencies usually below 100 kHz in both ECRH (central values $n_0 \approx 0.6 - 1 \times 10^{19}$ m^{-3}, $T_{e0} \approx 1$ keV) and NBI ($1 - 5 \times 10^{19}$ m^{-3}, 0.3 keV) plasmas; the other is magnetic-island-induced Alfvén waves with frequencies ranging from 50 to 300 kHz under NBI heating.

Island rotation modes are normally seen when the ideal low-order rational surfaces are located near the edge region ($0.5 < \rho < 0.9$, where ρ is the normalized minor radius) and there are two clearly distinguishable types characterized by stable or unstable rotation. Stable rotation happens in typical ECH and NBI L-mode plasmas, the latter characterized by peaked density profiles, with a frequency that seems compatible with ExB rotation. In H-mode plasmas the rotation speed increases but the mode dims noticeably as if the islands shrinked, eventually disappearing. Unstable rotation is a feature of the plasma at the threshold for the formation of a transport barrier, which is known to happen around the resonant layers [1]. The rotation is due to the same helicity as the stable mode but the frequency almost doubles in a short time scale (< 1 ms), compatible with having an electron diamagnetic rotation in the plasma frame [2]. The associated instability gives rise to all zonal-flow and predator-prey phenomenology described in TJ-II plasmas, which is indeed the threshold state characterized by repetitive bursts in magnetic activity well correlated with: bursts in H_α light, electron temperature flattening and drops in the diamagnetic energy and floating potential near the edge (measured with electric probes).

Alfvén waves below typical HAE frequencies have been investigated also in relation with the presence of islands [3]. Non-rotating islands add well-defined helicities to the B-field spectrum so frequency gaps can open for counter-propagating waves giving rise to weakly damped eigenmodes. The coupling between rotating island modes and Alfvén waves has also been detected: new modes with higher frequencies are excited suggesting a cascade of wave energy that channels fast-particle energy towards the thermal plasma.

In both cases, via rotation or coupling to Alfvén waves, low frequency modes associated to magnetic islands in plasmas of the TJ-II stellarator provide physical insights into plasma physics and, very importantly, possible means for transport and stability control.

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Studies of fast ion confinement in TJ-II in the presence of Alfvén waves

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The study of fast ion confinement in 3D devices like TJ-II is a key point for building a stellarator-based fusion reactor since alpha particles must be confined long enough to deposit their energy in the plasma bulk. Moreover, the fast ions coming from Neutral Beam Injection (NBI) or Ion Cyclotron Heating must be also well confined in order to increase the heating and current drive efficiencies. Fast ion transport must also be studied in order to find out what are the main escaping points of the ions, since they could damage the vacuum vessel of the device[1].
Fast ions can trigger instabilities [2], like Alfvén waves that can enhance transport through this kind of perturbations, which vary the magnetic field structure. The effect of Alfvén waves on fast ion transport must be added to the one driven by thermodynamical forces and orbits. The combined effect of NBI fast ion source and the 3D collisional transport produces a fast ion distribution function characterized by being non-monotonic. This kind of distribution functions is suitable to drive kinetic instabilities by inverse Landau damping. The instabilities caused by fast ions have been previously studied in the TJ-II stellarator[3], but no direct observation of the dependence on the ion beam energy of the induced ion transport has been reported so far.
In this work we study the fast ion confinement in the presence of Alfvenic instabilities in TJ-II stellarator by measuring the steady state fast ion population using the Compact Neutral Particle Analyzer (CNPA). We have found a clear decreasing of fast ion confinement when Alfvén modes are destabilized. The population of fast ions without AEs can be explained just by the orbit plus NBI source effect, as it is demonstrated by calculations with ISDEP code[4]. The resonant energy of ions with AEs is estimated from STELLGAP calculations and compared with the ones of missing ions. The fast ion confinement is studied under several experiments, including the modification of fast ion population by changing the current of NBI (in the range 44-60 A) as well as the energy of such ions, by varying the voltage of the injector (in the range 28-35 kV). The amplitude of the mode depends almost linearly with the population while it shows a weak dependence with the energy, for the range of energies explored here.

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Electromagnetic gyrokinetic simulations of Alfvén eigenmodes in stellarators

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One possible limit to the performance of a stellarator reactor is the loss of fast particles. Alfvén eigenmodes can be destabilised by interaction with fast particles, potentially resulting in greatly increased transport. In order to understand this behaviour in a possible future reactor, numerical models must be developed. In this work, we present the results of linear studies of Alfvén eigenmodes with fully global, electromagnetic, gyrokinetic models in W7X and LHD-like geometry. We also report on progress simulating such modes non-linearly. The ultimate goal of this work is to develop a realistic modelling capability for current and future stellarators.

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Topics: S3

MHD Stability of Edge Region of H-Mode Plasmas in LHD

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Clear suppression of m=1/n=1 edge MHD instabilities, of which likely candidate is resistive interchange mode (RIC), is observed in ELM-free phase of the H-mode plasma in a specific magnetic configuration of LHD where the low order rational surface ι=1 (ι: rotational transform) is placed just outside the last closed flux surface of the vacuum field. The edge transport barrier (ETB) having steep electron density gradient (accordingly steep pressure gradient) is formed in magnetic hill region, as shown in Fig.1. In the L-phase produced just after the H-L back transition, the m=1/n=1 RIC are enhanced strongly, although the edge pressure gradient is strongly reduced from that in the H-phase. Then, the RIC is again suppressed clearly in an improved L- phase (IL-phase) which is transiently realized just after the L-phase having an ETB structure in electron temperature profile not in density one, and has a similar character to I-mode [1]. The ELM free phase is interrupted by a large amplitude ELM induced by an explosive growth of RICs with an order of ~ 0.1ms time scale. If the ETB is formed just inside the ι=1 surface as inferred from Fig.1, considerable reduction of the pressure gradient at the rational surface is expected because of strongly reduced heat and particle transport. For this situation, considerably reduced linear growth rate of the m=1/n=1 RIC is predicted by a linear resistive MHD eigenvalue code [2]. When the ETB foot location expands beyond the ι=1 surface due to gradual increase of plasma stored energy in the ELM-free phase, the code predicts a strongly increased linear growth rate of the RIC, which may lead to larger growth of the RIC. This hypothesis qualitatively explain the RIC activities in the ELM mitigation by externally applied resonant magnetic perturbations (RMPs), since RMPs increase the pressure gradient at the ι=1 surface [3]. Diamagnetic and ExB flows may also play a role in the observed behaviours of RICs.

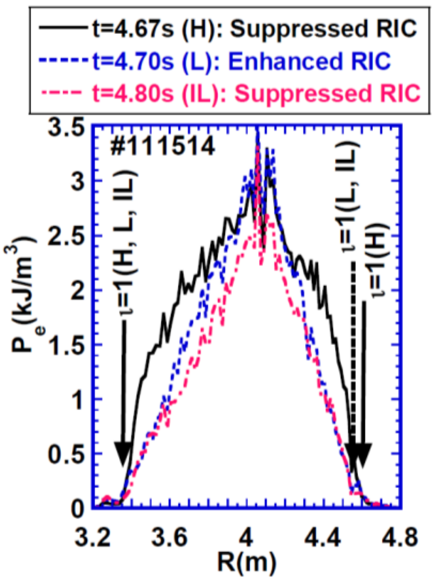

Fig.1 *Comparison of electron pressure profiles in ELM-free H, L and IL phases of an H-mode in the specific magnetic configuration. The vertical arrows indicate the inferred ι=1 surface for three time slices.*

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Estimation of internal structure of resistive interchange instability in LHD experiment

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Interchange instability is one of key issues for realization of thermonuclear fusion reactor because it would limit an achievable beta value and its gradient. Flattening structures in pressure profile observed in Large Helical Device (LHD) is considered to be driven by the interchange instability [1]. In order to understand the physical mechanism to limit the confinement performance by the instability, the accurate estimation of radial structure of the instability is required. According to previous works on the effect of the instability on the confinement property based on the radial displacement by multi-channel soft X-ray arrays, there is a clear dependence of degradation level for confinement property on peak value of radial displacement but the dependence on the radial width is not clear [2]. The reason for no dependence has two possibilities. One is the instability property. The other is the lack of measurement accuracy. In this research, the measurement accuracy of the instability's internal structure obtained by CO2 laser heterodyne imaging interferometer [3], which has higher spatial resolution and more channels than the soft X-ray arrays, is estimated in order to clarify the above reason. The line averaged electron density fluctuation δN_e normalized by the maximum value of DC component of the line averaged electron density signal N_{e0} is shown in a figure. When the profile of the instability is defined as $A\times\exp\{-((\rho-\rho_0)/w)2\}\times\cos(\theta+\phi+\omega t)$ in magnetic coordinate (ρ,θ,ϕ) with peak location ρ_0, peak value A and radial width w, these parameters is evaluated by comparison between the measurement and the model. In the discharge where an interchange instability with $\tilde{b}/B_t$ of 2.8×10^{-5} is excited, the profile with A of 0.38 %, w of 3.2 % and ρ_0 of 0.92 were obtained. According to the other results for different w, the interferometer can evaluate the w greater than 1% with accuracy of ∓1%. It is found that the interferometer has higher accuracy than soft X-ray arrays. That would enable the reason of the results obtained in the previous work [2] as shown in the above to be clear.

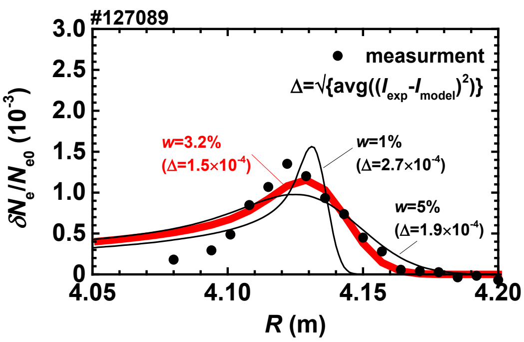

Figure. The internal structure of the interchange instability in LHD experiment.

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Visualization of low Coherent Fluctuation of Long Pulse Discharge Experiment in LHD

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In high temperature plasma, every fluctuations has possibility to couple another fluctuations and determines the whole plasma confinement. Fluctuations with low fluctuation level or low coherence, such as high-order harmonics of coherent mode or turbulence, would be no exception. Usually, such low coherence modes can be exposed by ensemble average of the spectra. Because the large number of spectra reduces the back ground level, experimentally, the long steady plasma is desirable to study such low coherence fluctuation.

In LHD, steady-state operation has been studied using electron cyclotron heating (ECH) and ion cyclotron heating (ICH) [1]. The typical discharge time is 1135 sec in condition of heating power 970kW. Because the density or temperature shifts basically slowly during long-pulse discharge, we can select long time windows (~100sec) for Fourier analysis. Figure 1 shows the comparison of coherence between the case of short (~0.4sec) and long (~100sec) time window. The observed plasma is sustained by ECH (370kW), and the centre of electron temperature is ~2.8keV. The fluctuation data of temperature measured by ECE is continuously acquired in rate of 500kHz-sampling. In case of the short time window case (number of spectra is 15), the back ground level is ~0.07. We see effective peak in only 1.13kHz and 90kHz. In case of long time window (number of spectra is 3051), we can clearly see not only the fundamental mode (1.13kHz) but also its 2nd harmonic 2.26kHz, and we can identify significant modes of 0.76kHz and 1.51kHz. In this way, the long pulse discharge in helical devices is useful for the study of behaviour of the low coherence fluctuation.In the presentation, the detail analysis of the long pulse discharge with magnetic axis sweep operation will be also discussed.

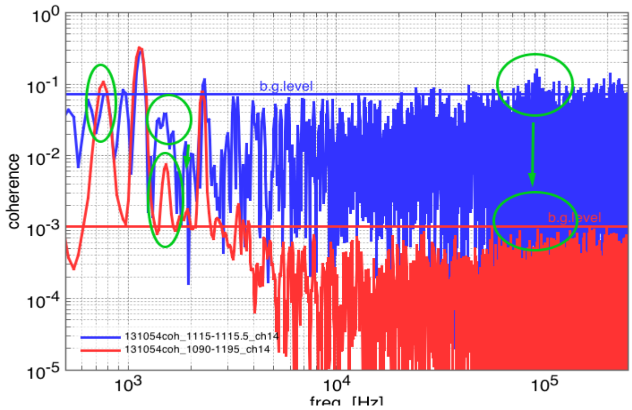

Fig.1 the comparison of coherence between short time window (0.4sec, 15 spectra, b.g.level 0.07) and long time window (100sec, 3051 spectra, b.g.level 0.001)

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Current Status of the Neutral Beam Injection System of W7-X

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Wendelstein 7-X (W7-X) will be equipped with a neutral beam injection (NBI) system. This system consists of two injector boxes with 4 independent beam sources each that are located equidistantly to both sides of the plasma bean shape symmetry plane in module 2 [1]. Dependent on ion source it is possible to inject between 2-2.5 MW of D^0 at 60 kV for 10 seconds pulse length or 1.5-1.8 MW of H^0 at 55 kV for 5 seconds pulse length. The system is planned to be commissioned in operational phase 1.2 of W7-X with a reduced configuration of two beam sources in each injector box [2].

The design of the NBI system closely follows the design of the ASDEX Upgrade injector II and was adapted only where necessary due to the specific geometric or operational requirements of W7-X or to further increase the reliability of the system. For example, due to the shape and location of the superconducting coils of W7-X the injection angle is almost perpendicular to the magnetic field. The small dimensions of the beam ports impose severe constraints on the alignment accuracy of the beam sources to avoid scrape-off losses. The close to perpendicular injection and imposes high power load requirements on the plasma vessel wall protection on the high field side. Further adaptations were necessary to the support structure of the beam boxes, the gas supply system, and the interface to the W7-X ports. The control, data acquisition and safety system is being developed from scratch according to the W7-X standards.

At present, design, manufacturing or procurement of most of the major components have been completed and assembly and integration of the system are ongoing. The support structure and the injector boxes have been installed in the experimental hall. The injector boxes have been aligned and connected via gate valves to the beam ducts of the plasma vessel. Most of the in-vessel components to protect the plasma vessel against beam losses have been installed. Since the magnetic field of the coils of W7-X is ramped up and down at most daily, more likely weekly, operation of the injectors must be performed with an ambient magnetic field. This required the development of an AC heating system for the high speed titanium getter pump system [3]. A fast IR based safety system is being designed to protect the high field side liner of the plasma vessel against shine-through. New solid state RF generators for the two of the beam sources are being procured.

The presentation will give an overview over the status of the: assembly, integration and commissioning. It will also provide information on physics issues to be addressed with NBI and possible restrictions during operation.

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Magnetic Field Optimization Study for Fast Ions Generated by ICRF Heating in Heliotron J

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Main purpose of this study is to optimize fast ion confinement and heating efficiency by using ICRF heating in a helical-axis heliotron device, Heliotron J (R_0 = 1.2 m, a = 0.1-0.2 m, $B_0 \leq$ 1.5 T) [1]. In Heliotron J, fast ion velocity distribution in the low density region has been investigated using fast protons generated by ICRF minority heating with special emphasis on the effect of the toroidal ripple of magnetic field strength (bumpiness) and heating position [2, 3]. The high bumpiness among three bumpiness configurations was found to be preferable for the fast ion confinement in the low density experiment and two dimensional dependence of fast ion distribution was experimentally investigated. The result of fast ion observation should be analysed including such effects and the configuration dependence of fast ion confinement and heating efficiency should be clarified.

Using Monte-Carlo method, the toroidal and poloidal dependence of the velocity distribution of minority hydrogen is calculated to comprehend the experimental results. In the toroidal dependence for the high bumpiness configuration, the high energy tail in the straight section is formed at 120° and 60° in pitch angle, however, only the tail at 120° is observed in the corner section. For the medium bumpiness, the toroidal dependence is very weak and the tail at 120° is dominant in every toroidal angle. For the low bumpiness configuration, the high-energy tail is smallest among three configurations and both peaks at 60° and 120° are observed. The high energy tail is larger in the outside area in the torus for every case. However, the change of energy spectra for various toroidal sections is little in each case.

To extend energy range of fast ions, target plasmas are produced by EC + NBI heating with 1×10^{19} m^{-3} in the line-averaged density. The injected hydrogen energy E_0 in NB is 24 keV. The experiment is performed in the high bumpiness and low-ε_t configurations. In the measured energy spectrum without ICRF pulse, E_0, $E_0/2$ and $E_0/3$ peaks are observed. When ICRF pulse is imposed, the tail is substantially increased, then, E_0, $E_0/2$ peaks are covered by the enhanced fast ion flux. Two configurations, the high bumpiness and the low-ε_t, are investigated. In low-ε_t configuration [4], enhancement of high energy tail is larger than that in the high bumpiness configuration. The energy of detected high-energy hydrogen by CX-NPA exceeds 50 keV near the injection pitch angle of NB in the low-ε_t configuration only. This configuration is considered to be most effective among the configurations examined for the ICRF minority heating in Heliotron J device.

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Effect of shape of resonance layer on acceleration process of ICRF minority ion in LHD

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In the Large helical device (LHD), a long time discharge is maintained for roughly one hour by the ICRF minority ion heating [1]. In the ICRF minority ion heating, the ICRF wave directly heats the minority ions near the resonance layer and then, the energy of the minority fast ions is transferred to bulk plasma through collisions. It is important for optimization of "an efficiency of transferred power from ICRF minority fast ions to bulk plasma" and "a ratio of an ion heating to an electron heating" to investigate the property of the velocity distribution of ICRF minority fast ions. The velocity distribution of the ICRF minority fast ions depends on the fast ion confinement, the acceleration performance of ICRF wave, and a collision frequency (slowing down process) with bulk ions and electrons. Particularly, it is important for estimation of velocity distribution of fast ions to investigate the effect of the shape of resonance layer on the acceleration performance of minority fast ions as well as the confinement of ICRF minority fast ions.

We investigate the effect of shape of resonance layer on a confinement property and acceleration performance of ICRF fast ions by using the ICRF fast ions analysis code [2], which uses simple model where the minority ions are accelerated only on the resonance layer. In LHD, a shape of resonance layer greatly changes with field strength. There are two major types of resonance layer shapes in LHD. The type which is a typical resonance layer in the LHD experiments is two resonance layers on the upper side and the lower side on a poloidal plan at the front of the ICRF antenna and the other type is one resonance layer at the front of the ICRF antenna like a tokamak. We will show the difference of the absorption power from ICRF wave and the average energy of the minority fast ions between two types of resonance layer shape. The ratio of the absorption power from ICRF wave to average energy of the minority fast ions is shown when magnetic configuration (fast ion confinement) and plasma parameter (slowing down process) is changed. The effect of the shape of resonance layer on the confinement property and the acceleration process of ICRF fast ions are discussed.

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Model Analysis of Plasma Start-Up by NBI with assistance of 2.45 GHz Microwaves in Heliotron J

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Neutral beam injection (NBI) plasma start-up assisted by non-resonant 2.45 GHz microwaves has been investigated in the Heliotron J device [1]. High power neutral beams are injected into seed plasmas produced by 2.45 GHz microwaves and then additional gases are puffed to build up high density plasmas. This scheme is strongly affected by the conditions of the seed plasma and by the gas puff timing. The threshold density of the seed plasma for high-density plasma build-up was observed around $1–3 \times 10^{17}$ m^{-3}. In order to investigate the threshold density and the physical processes such as a creation of fast hydrogen ion and an ionization of additional gas puff for Heliotron J, we have developed a zero-dimensional (0-D) model which is based on a NBI start-up model developed by Kaneko [2] and a start-up model using ohmic and electron resonance heating for tokamak devices [3, 4]. The 0-D model for NBI start-up is comprised of time dependent 9 equations for particle and energy balance: density equations of fast hydrogen ions (half and third energy components are included due to the positive ion source), density equations for bulk hydrogen and deuterium ions, density equations for hydrogen and deuterium atoms, and energy density for electrons and ions. The slowing down of fast hydrogen ions is included in this model since the slowing down time of fast hydrogen ions is shorter than the discharge duration. The calculations reproduce the threshold density of the seed plasma, the timing of rise in electron density observed in the experiment, and the dependence of plasma start-up on gas puff timing. In this meeting, effect of dissociation processes of hydrogen molecule will be also discussed.

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Impact of finite collisionality effects on electron cyclotron current drive in stellarators

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The study of electron cyclotron current drive requires the solution of the Spitzer problem, which is well known in the collisionless limit by the use of bounce average methods and in the high-collisionality regime where the classical Spitzer function can be applied. The drift kinetic equation solver NEO-2 [1] solves the Spitzer problem on a given flux surface for finite collisionality plasmas without simplifications on device geometry of tokamaks and stellarators using the adjoint approach [2]. This is performed by the method of field line tracing using the full linearized Coulomb collision operator including energy and momentum conservation, where in the current version the energy dependence is treated via expansion over test functions which are associated Laguerre polynomials of the order 3/2 (Sonine polynomials) [3]. Recently, precomputed results of the parallel version of the solver NEO-2 were integrated into the ray-tracing code TRAVIS [4] by an efficient interface in order to investigate the impact of finite collisionality effects on current drive in Wendelstein 7-X.

This report shows that the correct treatment of collisional effects leads to an increase of the generated current drive, especially for particles close to the trapped-passing boundary, compared to results obtained in the collisionless limit computed by the internal TRAVIS model and the bounce average solver SYNCH [5]. In addition, a more flexible description of the energy dependence of the collision model is introduced using an expansion over an arbitrary complete set of test functions. These test functions are well suited to represent the form of the solution in velocity ranges of interest, especially in velocity regions important to ECCD modeling.

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Control and Reconstruction of 3D equilibria in the MST RFP

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Control of the orientation of 3D equilibria in the MST using an m=1 RMP has recently been achieved [1]. This has led to greatly improved diagnosis, revealing enhancements in the central density and temperature. Coupled with a recent advance in V3FIT-VMEC, reconstructions of the 3D equilibria have also been dramatically improved. The RMP also inhibits the generation of high-energy ~50 keV electrons that is otherwise common with the 3D state (Fig.1). This state occurs in sufficiently hot RFP plasmas when the innermost resonant m=1 tearing mode grows to large amplitude (Fig.1), producing a helical magnetic axis. As the mode grows, eddy current in MST's conducting shell slows the mode's rotation. This leads to locking of the resultant 3D structure but with a randomly varying orientation shot to shot. An m=1 RMP can now be applied with a poloidal array of saddle coils mounted on the insulated cut in the shell. With an amplitude b_r/B~10% and a tailored temporal waveform, the RMP can force the 3D structure into any desired orientation relative to MST's diagnostics. A recent advance in V3FIT-VMEC allows calculation of the substantial helical image current flowing in MST's shell, which has in turn allowed self-consistent utilization of both external and internal (laser Faraday rotation) measurements of the magnetic field. This reflects MST's continuing contributions to the testing and development of this important 3D reconstruction tool. The ORBIT code predicts reduced magnetic stochasticity and improved confinement of high energy electrons within the 3D structure. The suppression of these electrons by the m=1 RMP may reflect a change to the central magnetic topology. However the generation of these electrons is unaffected by non-resonant perturbations, such as m=3. The success of the RMP technique also bolsters the helical island divertor concept for the RFP, which in analogy to stellarators is one possible future means for power and particle handling, but which depends on control of the 3D equilibrium orientation.

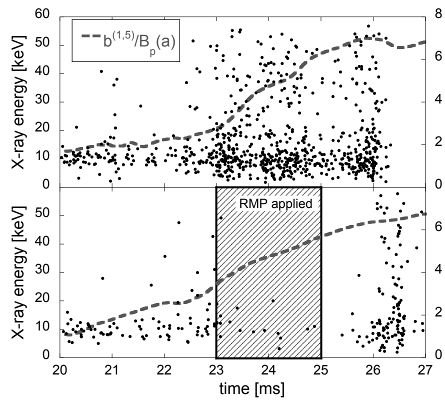

Figure 1: X rays flux from high energy electrons without (top) and with (bottom) an RMP applied near the time when the m = 1, n = 5 mode grows to large amplitude.

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CKA-EUTERPE: A kinetic MHD model for nonlinear wave particle interaction

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Energetic particles in a fusion reactor which may originate from either the fusion itself or NBI injection can interact with waves in the plasma and drive them unstable. The resulting fields may lead to an increased loss of energetic particles endangering plasma performance and reactor walls.

Based on the completely gyro-kinetic EUTERPE code, an MHD kinetic hybrid code CKA-EUPERPE had been developed earlier which used the mode structure obtained from solving the reduced MHD equations to compute the power transfer between the mode and the particles to infer the growth rate.

This model is extended to work in a non-linear regime. Still, the spatial structure of the MHD mode is kept fixed but the amplitude and phase are allowed to change in time according to the calculated energy transfer from the fast particle component.

The necessary equations are derived from the gyro kinetic equations. As the $v_{||}$-formalism is closer to the physical picture, the code was changed to the $v_{||}$-representation.

The nonlinear saturation of modes is investigated in two and three dimensional geometries. For tokamaks, using the settings of the ITPA benchmark case, the results will be compared with those of other codes. It will be investigated how a finite gyro radius of the fast particles alters the saturation behaviour.

Linear and non linear calculations will be done in stellarator geometry, especially for the HELIAS reactor using alpha particle drive. The saturation level is compared with that of tokamaks.

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Global m=n modes and their destabilization in forthcoming NBI experiments on Wendelstein 7-X

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Plasma instabilities driven by energetic ions, which have different features and consequences, were observed in many toroidal fusion facilities. In particular, instabilities excited by the ion beams produced during plasma heating by Neutral Beam Injection (NBI) were observed in the Wendelstein 7-AS (W7-AS) stellarator [1-3]. In this device beam-driven instabilities occurred in both low-beta discharges and high-beta discharges; however, in the latter case they were transient and occurred after switching to full NBI power and during density ramp up. It is of importance to see what will be the influence of energetic ions on the plasma stability in Wendelstein 7-X (W7-X), the optimized stellarator of the same line. This motivated us to carry out this work aimed to contribute to the study of plasma stability during NBI in W7-X.

Wendelstein 7-X, like its predecessor Wendelstein 7-AS, is characterized by a small magnetic shear. However, the rotational transform of the magnetic field lines, ι, in W7-X, in contrast to that in W7-AS, is close to unity. Note that ι in the Helias reactor (the Wendelstein-line machine) will be close to unity, too [4].

It is shown in this work that modes of Alfvénic character with global radial structure exist and interact resonantly with the injected energetic ions, which are destabilizing. They have equal and small poloidal and toroidal mode numbers $(m = n)$ and can be excited when $2 \le m << 10$.

The analysis includes (i) derivation of an equation for the scalar potential of the perturbed electromagnetic field in a compressible plasma containing energetic ions, (ii) calculation of eigenmodes using this equation and the CAS3-D code, (iii) both perturbative and non-perturbative stability analysis, (iv) consideration of the characteristics of injected ions and effects on them of the radial electric field. The high-mirror magnetic configuration was employed. It was assumed that NBI produces ions with the maximum energy of 55 keV when hydrogen atoms are injected and 60 keV in the case of deuterium injection, with pitch angles and the total power equal to that planned on the first W7-X experiments with NBI.

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Fluctuations in the Alfvén Range of Frequencies in the H-1NF Heliac

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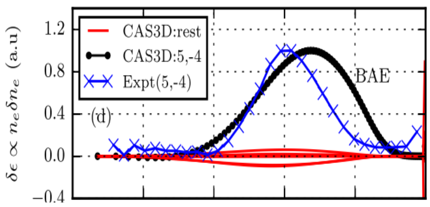
Despite its modest parameters, the H-1 heliac is a remarkably good test bed for fundamental understanding of modes on the shear Alfven range of frequencies. The plasma beta is high enough that sound mode interaction is significant and the strong three dimensional effects present challenges representative of advanced geometries such as the W-7X, device to be commissioned this year. Furthermore, the wide range of magnetic configurations accessible to H-1 makes it an ideal device for studying the parametric dependence of these instabilities which are observed in the low to mid Alfvénic range in H/He RF produced plasma around 0.5T, in the range 5-300kHz. Scans of plasma composition and magnetic field enabled by recent RF upgrades, tomographic determination of fluctuation profiles, high resolution background density profiles and new datamining techniques have shed new light on both the magnetic and density fluctuations. More than 80 magnetic probes in three arrays, and several optical imaging diagnostics provide data on poloidal, helical and radial mode structures. Recently all three components of the 16 element helical magnetic probe array have been analysed and provide further information on the nature and polarisation of the modes. A new sensitive multi channel interferometer, and framing cameras synchronised with the mode, provide density fluctuation data in two dimensions, covered in more detail in [1]. Together these provide a picture of the radial and poloidal mode structure, its polarisation and its dependence on rotational transform.

The vast data set acquired shows very clear dependences on transform and density, made possible by the low shear and precise control of configuration available in H-1. Magnetic field scans indicate Alfvénic scaling in some cases, and less dependence on B in others. Both Alfvénic GAE-like and/or acoustic-like (BAE) behaviour [2] is seen in different regimes and the relationship of the observed modes to GAMs and acoustic modes is examined. In particular density fluctuation profiles support the BAE interpretation (see figure). Interpretation is discussed in relation to predictions of CAS3D, CONTI and preliminary damping and drive estimates. Some challenges presented by strong shaping and finite beta will be considered. Finally the various datamining techniques that have been applied to H-1 will be briefly reviewed and their prospects for providing further understanding assessed.



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Three-dimensional numerical analysis of shear flow effects on MHD stability in LHD plasmas

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In the Large Helical Device (LHD), experiments for high beta achievement has been extensively conducted. The highest value of $\langle\beta\rangle = 5.1\%$ was obtained in the configuration with $R_{ax} = 3.6$m and $\gamma_c = 1.20$ [1]. Here, R_{ax} and γ_c are the horizontal position of the vacuum magnetic axis and the parameter of the aspect ratio of the helical coils, respectively. On the other hand, the plasma in this configuration is predicted to be unstable with respect to the Mercier criterion. It follows that some stabilizing effects work on the LHD plasma for the achievement of the high beta. In order to design the coil configuration in the heliotron DEMO accurately, it is crucial to identify the stabilizing mechanism in the viewpoint of the magnetohydrodynamic (MHD) stability. In the recent experiments, the phenomenon like a locked mode was observed [2,3]. In this case, the mode grows rapidly just after the mode rotation stops, and brings a partial collapse of the profile in the electron temperature. This phenomenon indicates that the shear flow of the plasma may be a candidate which can suppress the growth of the mode. Thus, we numerically study the effects of the shear flow on the stability against the pressure driven modes in the LHD plasmas. For this purpose, we utilize three-dimensional (3D) numerical codes. It has not been established how to obtain a 3D equilibrium consistent with finite flows. Hence, we utilize a static equilibrium, and then, we set a finite flow in the stability examination, as the first step of the flow analysis. In the equilibrium and the stability calculations, the HINT2 code [4] and the MIPS code [5] are utilized, respectively. In both codes, rectangular grids are employed in the coordinates (R, ϕ, Z). The MIPS code examines the stability by following the time evolution of the plasma. We set the shear flow $\boldsymbol{v}$ satisfying $\boldsymbol{v} \cdot \nabla\Psi = 0$ as the initial condition of the MIPS code. Here, Ψ denotes the label of the equilibrium flux surface. The profile of the flow close to the experimental data is assumed. It is reported how the shear flow affects the linear growth rate and the nonlinear dynamics of the pressure driven modes.

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Bifurcation of the interchange mode growth rate and rotation frequency due to the perpendicular heat conductivity in stellarator plasmas

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The linear stability against interchange modes of heliotrons like the Large Helical Device has not been fully understood yet. The interchange mode is the dominant instability in nearly current-free 3D configurations. Recently, experiments in the so-called inward-shifted configuration have shown the machine can be operated safely up to $\langle\beta\rangle \sim 5\%$ without major MHD event [1]. Here $\langle\beta\rangle$ is the volume averaged ratio between kinetic and magnetic pressure. The rotation of the mode, which seems to play a role in the stability [2], is also not understood. Namely, the modes are observed to rotate in the electron direction [3], whereas extended MHD theory without heat conductivity but including electron and ion diamagnetic effects predicts rotation in the ion direction for Mercier unstable modes and almost no rotation for Mercier stable modes. [4].

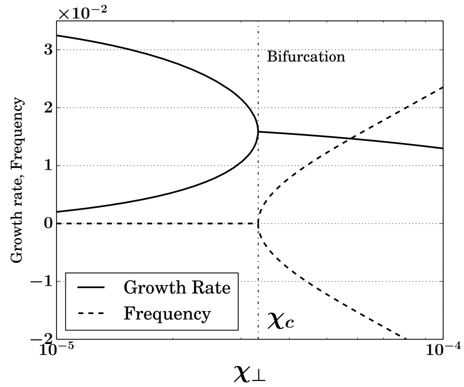

Figure 1: Frequency and growth rate of the first two most unstable modes

In this contribution, we present a new effect on the linear Mercier unstable interchange growth rate caused by the perpendicular heat conductivity $\chi_\perp$. When $\chi_\perp$ is equal to a critical value χ_c, the growth rates of the two most unstable modes become equal. For $\chi_\perp > \chi_c$, the two modes have same growth rate and opposite frequency, as shown in Fig. 1. This degeneracy is removed by the diamagnetic effects. In the domain $\chi_\perp > \chi_c$, the diamagnetic effects can be destabilizing and the rotation can be in the electron direction. The rotation direction is determined by the value of perpendicular viscosity.

This effect, which sheds a new light on the stability of the interchange mode in stellarators, is found analytically and confirmed both with a 3 field linear eigenvalue solver and with 3D non-linear MHD simulations utilizing the MIPS code [5].

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Optimizing Stellarators for Energetic Particle Confinement using BEAMS3D

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Energetic particle (EP) loss has been called the "Achilles heel of stellarators,"[1] and there is a great need for magnetic configurations with improved EP confinement.

In this study we will utilize a newly developed capability of the stellarator optimization code STELLOPT: the ability to optimize EP confinement via an interface with guiding center code BEAMS3D[2]. Using this new tool, optimizations of the W7-X experiment and ARIES-CS reactor are performed where the EP loss fraction is one of many target functions to be minimized.

In ARIES-CS, we launch 3.5 MeV alpha particles from a near-axis flux surface. STELLOPT launches particles from a user-defined toroidal and poloidal grid, with user-defined pitch angles. As these particles are born from D-T reactions, we consider an isotropic distribution in velocity space. In W7-X, we simulate the experimental NBI system using realistic beam geometry and beam deposition physics. The goal is to find configurations with improved neutral beam deposition and energetic particle confinement.

Previous studies of this type have been conducted using geometric proxies thought to correlate with EP losses[3]. In addition, Mynick et al.[4] optimized a reactor-sized NCSX configuration using the ORBIT3D code, but only followed alpha particles for a small fraction of their slowing-down time. We will discuss our results in the context of these previous studies, and suggest paths for future research.

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Further Extensions of Development of Integrated Transport Analysis Suite, TASK3D-a, and Applications to LHD Experiment

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The integrated transport analysis suite, TASK3D-a (its first version: a01, upper part of Fig. 1 in which calculation flow is shown), has been developed and applied mainly to NBI-heated LHD plasmas [1]. Recently, further extension has been conducted such as including ECH ray-tracing codes (TRAVIS [2] and LHDGauss [3]) and the module for creating ascii files to be registered onto the International Stellarator-Heliotron Confinement Database [4,5]. Inclusion of ECH ray-tracing code can significantly enhance systematic energy transport analysis of ECH- (and NBI-) heated LHD plasmas, for which previously stand-alone ECH absorption calculations has been performed such as reported in Ref. [6]. Further extensions for physics analyses such as on the plasma flow, Alfven eigenmodes, energetic particles, and others, have been progressing. Details will be reported in the workshop.

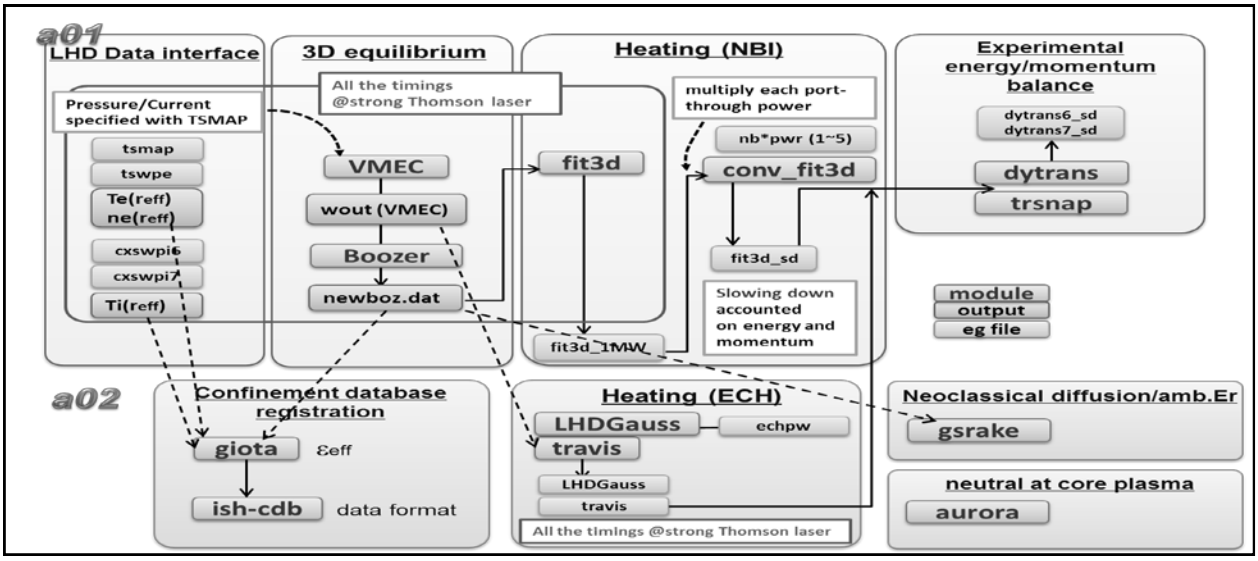

Fig.1: Calculation flows in TASK3D-a01 (upper part) and a02 (being extended).

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Wave-particle interaction analyser for study of Alfven eigenmodes in the Large Helical Device

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Recently, stabilization of energetic-ion-driven Alfven eigenmodes (AEs) due to the ECH application has been observed experimentally [1-3] and attracts much attention from the viewpoint of the possibility of external control of AEs. However, no model can explain the stabilization of AEs due to ECH application [3], and systematic experimental studies are necessary to understand the physic mechanism. Here, we propose to apply the wave-particle interaction analyser (WPIA) technique being developed for magnetosphere plasma physics (ERG project) to magnetically confinement fusion experiments for study of AE stability.

The concept of WPIA is a phase detection between particle flux and the wave for the quantitative evaluation of energy transfer between them. We have developed a high speed pulse analyser system for WPIA using the field programmable gate array (FPGA) module, and installed the system to the large helical device (LHD). One channel of Mirnov signal and eight channels of semi-conductor fast neutral analyser (Si-FNA) signals are digitized with sampling rate of 50MS/s (maximum), which is significantly higher (factor of 10^4) than that of conventional pulse height analyser and enable us to evaluate the phase with respect to the wave.

In the conference, the AE stabilization due to ECH application in LHD will be presented, and the some phase space structures observed by a WPIA system installed in LHD will be also discussed.

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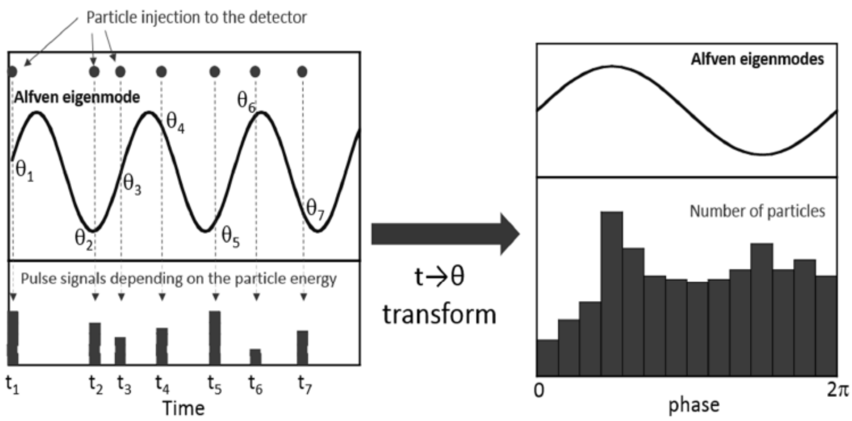

Fig. 1 Conceptual drawing of wave-particle interaction analyser (WPIA).

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Real space and flux coordinate calculations of fast particle losses in the optimized stellarator.

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Minimization of fast particle losses is one of the main targets of stellarator optimization. Such computation are typically performed in process of optimization using the magnetic field representation in flux coordinates where the assumption of embedded flux surfaces is used. This assumption may lead to solutions where the equilibrium equations are satisfied only approximately, and in this case representation of the magnetic field in real space coordinates may provide a different result for the confinement properties of the device as compared to flux coordinates.
In this report the results on fast particle losses computed for the optimized stellarator configuration Hydra-23 [1] using the magnetic field representation in flux (Boozer) and in real space (cylindrical) coordinates are compared. In addition, the result for the effective helical ripple and simplified semi-analytical criteria characterizing the fast particle confinement are also presented.

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Session 4

Impurity, neutral sources, sinks and transport

THREE-DIMENSIONAL SCRAPE OFF LAYER TRANSPORT IN THE HELICAL SYMMETRIC EXPERIMENT HSX*

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In this work, experimental characterization of the edge region of the Helically Symmetric Experiment (HSX) is presented. The edge topology of HSX in the quasihelically symmetric configuration is characterized by an 8/7 magnetic island remnant embedded in a short connection length scrape-off layer (SOL) domain. The goal of this research is to understand the role of the complex magnetic topology structure on the heat and particle transport as well as neutral and impurity sourcing. Detailed comparisons of the measurements to edge simulations [1] using the EMC3-EIRENE [2] 3-D plasma fluid and kinetic neutral gas transport model are performed.

The plasma structure and related transport in this anisotropic, three-dimensional (3D) plasma boundary at HSX was investigated for the first time using a multi-pin, flapping Langmuir. This diagnostic provides two-dimensional profiles of the electron temperature, electron density, parallel ion flows, and floating potential.

The measurements show that particle transport is diffusive within the island region and dominantly convective in the SOL region with evidence for cross-field diffusion coefficients in the SOL that are very low ($D \sim 0.03$ m^2/s). In contrast, the theory/simulation comparison suggests that due to low particle transport, the heat transport is governed by radial heat diffusion across the SOL domain that exceeds convective transport. Parallel flow structures are measured in the HSX edge, with strong flows near the island X-point. The flow pattern and magnitude is not in agreement with EMC3 EIRENE modeling, pointing towards uncertainties in the actual magnetic field topology or additional flow drivers not accounted for in the model. A prominent feature of the measurements is the presence of a positive plasma potential structure centered near the island O-point. The potential can drive ExB flows in the interface region between the edge island and the last closed flux surface which are not yet included in the EMC3- EIRENE modeling.

The impact of the flows driven by parallel pressure gradients and of the ExB flows on SOL transport will be discussed including the link to the plasma source/sink relation. These results, based on the map of plasma parameters from the Langmuir probe provide a detailed overview of 3D scrape-off layer transport in HSX, and can inform results from low shear stellarators with island divertors like W7-X.

* This work supported by US DOE Grant DE-FG02-93ER54222 and DE-SC0006103

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Particle source and edge confinement study based on spectroscopic diagnosis in the LHD

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In recent LHD research much attention has been drawn by a detailed analysis of the Hα line profile which has demonstrated its ability to give the neutral penetration profile from edge through core in the plasma. This analysis is based on an insight that atoms existing in the plasma would have the same temperature as protons at the same location because they may be dominantly created from protons by the charge exchange process. The ionization rate, or the particle source rate, and the atom density in the entire region of the plasma are derived by the Laplace inversion of the measured line profile [1, 2].

The particle confinement time τ_p of the volume inside a given magnetic surface can be evaluated with the obtained particle source rate profile. Figure 1 shows the results as a function of the magnetic surface location designated by the effective minor radius r_{eff} for two discharges with different magnetic field strength. It is clearly seen that τ_p with B_{ax} = 2.75 T is approximately one order of magnitude larger than that with B_{ax} = 0.41 T, where B_{ax} is the magnetic field strength at the magnetic axis. In addition, a sudden change in the gradient is seen for B_{ax} = 2.75 T at around r_{eff} = 0.6 m, which corresponds approximately to the location of the last closed flux surface. The reason for the obscure boundary in terms of τ_p for B_{ax} = 0.41 T is still open. We have also applied this methodology to discharges with the resonant magnetic perturbation (RMP) and the influence of RMP on the edge confinement characteristics is investigated in detail.

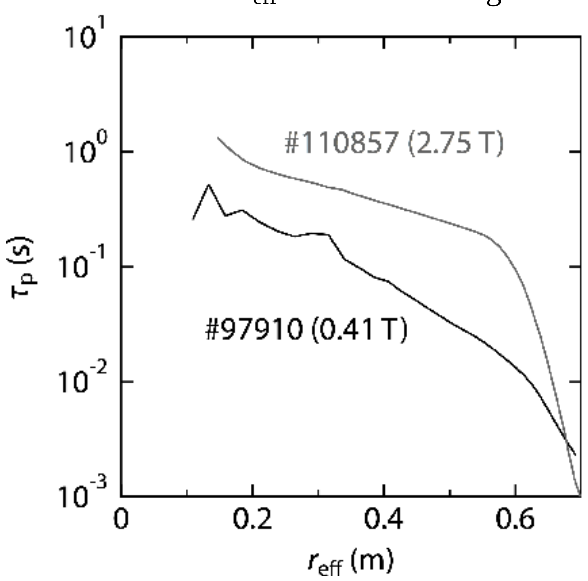

Fig. 1: Dependence of τ_p on the magnetic surface position designated by r_{eff} for two discharges with different magnetic field strength.

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Impurity density variation on flux surfaces in stellarators

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Attaining predictive capability and active control of impurity transport is necessary to ensure a stationary plasma operation with tolerable heat loads on plasma facing elements. For stellarators, the consistently negative ambipolar radial electric field, E_r, is predicted to cause impurity convection towards the core. Yet, there are a number of experimental conditions in which such accumulation is not observed even if E_r is measured or predicted to be negative [1, 2]. The hollow impurity core profiles observed in LHD require a fundamental change in the radial convection of impurities, the cause of which has not been identified to date. Turbulent radial flux [3] and electrostatic potential variations on flux surfaces [4] have recently been inspected from a theoretical perspective as possible candidates to cause outward convection in ion-root ($E_r < 0$) plasmas. The latter can grow particularly large in stellarators because of unfavourable scaling of the corrections to the Maxwellian ion distribution function with collisionality and its interplay with impurity density variations can significantly alter radial impurity fluxes.

In this talk we will present a theoretical revision of the mechanisms that can give rise to large inhomogeneities in the density of impurities and electrostatic potential and will discuss the implications for radial impurity transport. Theoretical expectations and simulations will be confronted with experimental observations of asymmetries in TJ-II [5, 6], recently complemented with heavy ion beam probe data and tomographic reconstructions of bolometer arrays in pellet injection experiments. The status of our understanding of the experimental observations will be summarized.

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Relation between impurity/hydrogen emissions and edge stochastic magnetic structure at detachment transition observed in LHD divertor region

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A measurement system of visible spectroscopy viewing the divertor region of LHD has been developed to study the relations between impurity/hydrogen emissions and the edge stochastic magnetic field structure. The system consists of a 2D array (10 x 13 = 130 channels) of optic fibers and of a spectrometer with measurement wavelength range of 200 – 1000 nm. The viewing area covers the divertor plates, divertor legs, edge stochastic layers, the last closed flux surface (LCFS) and the confinement region. The system provides a 2D profile measurement of impurity/hydrogen emission spectra at the divertor region, which enables us to characterize the detached divertor plasmas in relation to the edge magnetic field structure.

At the detachment transition the emissions from carbon (as a divertor material in LHD), CII, CIII, CIV, and from hydrogen H_β, H_γ, have been measured. The following observations were obtained:

1. The carbon radiation is peaked around the X-point of divertor legs rather than at the divertor plates even before the detachment transition, where the detachment transition is defined as a reduction of divertor particle flux.
2. The CIV (772.6 nm, n=6-7), which is considered to originate from CX between neutral hydrogen and C^{4+} or recombination of C^{4+} with electrons, increases significantly after the detachment transition around the X-point. At the same time, H_β also peaked at the X-point. The results indicate formation of cold and dense plasma at the X-point during detachment.
3. The ratio of H_γ to H_β, as an index for hydrogen volume recombination, increased along the LCFS after detachment transition.
4. The Doppler shift analysis of CIII provides 2D flow structure, which shows impurity flow alternation pattern according to the different field line directions connecting to the divertor plates. The flows towards divertor plates were estimated at of order of 10^4 m/s. After the detachment transition, the flow pattern changes and the velocity tends to decrease.

At the conference, the results of auxiliary impurity puff experiments, e.g. Kr as a high Z impurity, Ne etc., will be also presented.

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Enhancement of Helium Exhaust by Resonant Magnetic Perturbations at LHD and TEXTOR-DED

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Exhaust of helium as a fusion born plasma impurity is a critical requirement for future burning plasmas. The effective total helium confinement time $\tau_{p,He}$* must not exceed the energy confinement time τ_E by more than an order of magnitude to avoid dilution of the fusion fuel.

We demonstrate in this paper that resonant magnetic perturbation (RMP) fields can be used to actively manipulate helium exhaust features. We present results from TEXTOR-DED as example for a tokamak with a pumped limiter and from the Large Helical Device LHD with the closed helical divertor as example for a heliotron device. In both devices RMP fields are applied to generate a magnetic island located in the very plasma edge. This magnetic island has a noticeable impact on the helium confinement in both devices. Without RMP field applied, $\tau_{p,He}$* is a factor of 3.5 higher for LHD compared to TEXTOR discharges in a comparable plasma density range. Ion root transport - one out of several different impurity transport regimes at LHD - is the most likely inward transport driver causing the high $\tau_{p,He}$* values in this transport regime. However, $\tau_{p,He}$* is decreased by up to 30% and hence values in tendency closer to the tokamak situation can be established with an RMP induced edge magnetic island.

The measurements from both devices support that this reduction in $\tau_{p,He}$* is caused by combination of improved helium exhaust due to enhanced outward transport, improved coupling to the pumping systems and reduced fueling efficiency for injected and recycled helium. A radially resolved analysis of the He/H ratio for LHD shows that enhanced helium exhaust is particularly significant at the resonance layer where the island forms and outside in the remaining chaotic edge surface layer. This points out the need to consider the island region and both, the region inside and outside as separate particle reservoirs for more complete particle balance analysis. Such a multi-reservoir particle balance will be applied to better understand inward and outward helium transport in direct relation to experiment. EMC3-EIRENE analysis of the edge plasma domain for both devices will be discussed to resolve the strong role of the edge magnetic island for helium exhaust.

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Helium Transport and Exhaust on LHD with EMC3-EIRENE

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We investigate helium transport and exhaust in a puff-pump experiment. A proper understanding of helium transport is necessary for fusion reactor scenarios where exhaust of helium ash is a necessity for operation.

For a case study, we present simulation results using EMC3-EIRENE of LHD (Large Helical Device) equilibrium plasmas including an impurity helium puff. The simulations are compared to recently obtained experimental data from LHD. Configurations with RMP (Resonant Magnetic Perturbation) coils and without RMP coils are presented. The RMP coils produce a prominent 1/1 island in the edge of the plasma. In both configurations, a helium puff is included in the plasma simulation along with the background carbon impurity.

Experimental data show that the system confinement time, τ_p^*, of helium in the edge layer is reduced when the RMP fields are implied. This is visible spectroscopically by a strong depletion of helium at and outside the 1/1 island. This hints to increased helium transport most likely caused by the presence of the 1/1 island. If the reduced τ_p^* of helium is a result of the different magnetic geometries and the resulting differences in plasma densities and temperatures, then the EMC3-EIRENE simulation will capture this phenomenon. Alternatively, if the shielding results from differences in anomalous transport or other effects such as an electric potential in the 1/1 island, then the simulation will not see a significant difference in the helium dynamics in the LHD edge. From the simulation we can begin to understand the key physics issues that govern helium transport in 3D edges.

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Mitigation of Core Impurity Accumulation by EC Heating in LHD*

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In a magnetic confinement fusion device, various impurities from low-Z material e.g. a helium ash, which is a by-product of a fusion reaction, to high-Z one e.g. tungsten, which is derived from a possible plasma facing component, will exist inside the plasma. When the amount of the impurities exceeds the acceptable level for some reason, e.g. due to the accumulation, it can cause a significant fusion reactor performance degradation. Therefore, it is crucially important to develop an effective scheme for controlling the amount of the impurities in the core plasma, especially for removing the impurities from the core plasma. In this contribution, we show a demonstration of the drastic mitigation of the core impurity accumulation by applying an additional electron cyclotron heating (ECH) in a helical plasma. The demonstration experiment has been performed in LHD, where the impurity accumulation has been observed in a high-density discharge [1]. As shown in Fig. 1, a tracer impurity, vanadium (V, Z = 23) is externally injected (here at t = 3.95 s) by using the TESPEL technology [2], and then the collisionality between the tracer impurity and the bulk ion in the core plasma will be in the PS regime. The plasma without the additional ECH shows clearly a strong accumulation and then it causes the reduction in core T_e, followed by the decrease in the intensity of the V Be-like line emission. This decrease could be attributed to the modification of the charge state distribution of the vanadium ion by the drop in core T_e. Meanwhile, when the 154 GHz ECH (P_{ECH} ~ 1.5 MW, ρ_{abs} < 0.5) was applied just after the TESPEL injection, the accumulation of the tracer impurity was almost completely mitigated. Even after the 154GHz ECH was switched-off, the intensity of V Be-like line emission was increased only slightly. In this contribution, the impurity transport analysis result with STRHAL will be also discussed for clarifying the essence of the ECH effect on the impurity transport in helical plasmas.

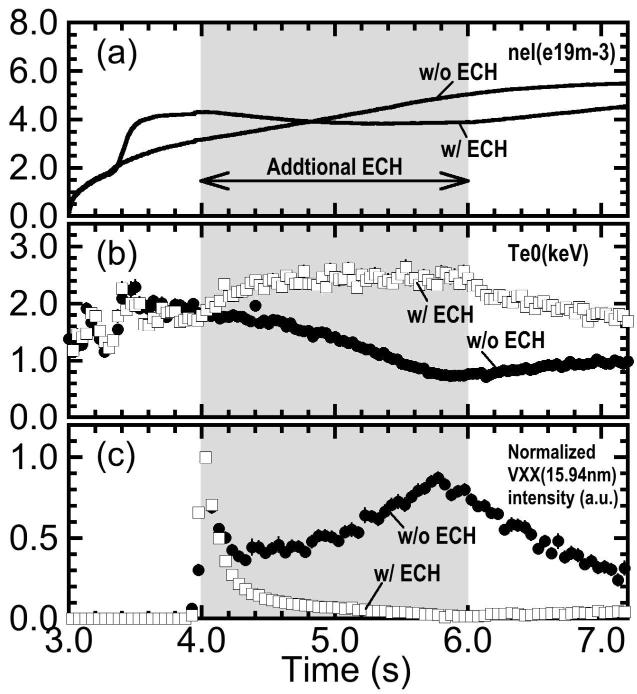

Fig. 1 Demonstration discharge of the mitigation of core impurity accumulation by applying the additional ECH (from t = 4.0 s to t = 6.0 s, P_{ECH} = 1.5 MW, ρ_{abs} < 0.5). As a guide, waveforms without applying the additional ECH are also shown. The tracer impurity, vanadium was injected at t = 3.95 s by using the TESPEL technology.

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Numerical Simulation of the Active Beam Spectroscopy on W7-X

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In the new stellarator Wendelstein 7-X (W7-X) a couple of active beam assisted spectroscopy diagnostics will be realized. This will be the Charge eXchange Resonance Spectroscopy (CXRS) and Beam Emission Spectroscopy (BES) as well as the spectroscopic measurement of the Motional Stark Effect (MSE). One of the neutral beams will be provided by a Russian Diagnostic neutral beam Injector (RuDI-X) with low injected power (typ. < 200 kW) and low divergence to improve the spatial resolution for the measurement of impurity densities, the ion temperature and the radial electric field. The other neutral beams are delivered by the heating Neutral Beam Injection (NBI) system, optimized to high power feed-through into the W7-X plasma (up to ≈ 10 MW during H_2 operation). A MATLAB code had been written in order to simulate in a predictive manner the results of the planned active spectroscopy systems in the neutral beams mentioned above. As fundamental input for the simulations, the CATIA data of the W7-X machine hardware are used, as well as the geometrical and optical layout of the spectroscopic diagnostics, i.e. the positioning of plasma facing windows, imaging optics, the quartz fibre arrays, the spectrometers and the CCD-cameras behind. The location of the spectroscopic measurements in the plasma, i.e. the mapping between spatial and magnetic co-ordinates is handled by fetching pre-calculated VMEC equilibria. Background plasma parameter profiles, like the electron and impurity densities, ion and electron temperatures and plasma rotation velocities, are so-far available as best-guess models, based on neoclassical transport calculations. For the estimate of the impurity density profiles, 1-D impurity transport calculations are performed with the STRAHL code. For the input of atomic data, the ADAS database is used. The MATLAB code calculates the neutral beam absorption (power deposition) in the plasma for the three energetic components, and uses a heuristic model for the slowing down of the energetic ions. The excitation of CX active lines is calculated for a couple of impurities (He, B, C, Ar, Ne), and the measurement of the CX line profiles is evaluated, including simulated noise during the measurement process. The calculation of CX spectra on a fast He^{++} ion population (for instance after He beam injection into W7-X) is also included to assess the possibility of He-ash studies and fast ion confinement in W7-X. The motional Stark pattern of the beam neutrals is calculated in the W7-X magnetic field to assess in advance the properties of the MSE measurements, including the impact of the bootstrap current which is decisive for the magnetic plasma edge configuration, i.e. the island divertor performance. Finally, Neutral Particle Analyzer (NPA) spectra are calculated for thermal and high energetic CX neutrals leaving the plasma.

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WENDELSTEIN 7-X IN THE EUROPEAN ROADMAP TO FUSION ELECTRICITY

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The Roadmap to fusion electricity is the European plan to demonstrate fusion electricity by 2050. The plan is implemented by the Consortium of European Fusion laboratories (EUROfusion). The Roadmap consists of eight missions, all of which address key elements en-route to DEMO and follow a success oriented mitigation of potential development risks by the implementation of appropriate countermeasures and the development of key technologies.

Stellarators are an integral part of the Roadmap. The specific stellarator mission explores the most promising alternative line to the tokamak and aims at bringing the stellarator line to maturity. The objective of W7-X in the line with Helical Advanced Stellarators (HELIAS) is to demonstrate this advanced stellarator concept to be an attractive candidate for reactor extrapolation.

The EUROfusion program is planned to be an integral part in the first experiment phases of W7-X, contributing by diagnostics and heating components, modeling and the preparation of experimental schemes making benefit from the smaller, but easier-to-access device TJ-II. The program is both to participate in the physics campaigns of W7-X and by the preparation of strategic elements in view of the Roadmap.

In the first limiter campaign (OP1.1) of W7-X, the physics contributions cover electron-root studies, safe operation and breakdown with ECRH, impurity monitoring and SOL studies. These topics are further evolved in the later campaigns (OP1.2). OP 1.2 is understood to prepare key elements in view of steady-state and high-density operation even in shorter pulses when the device is still equipped with an uncooled divertor (e.g. physics of exhaust, heating (ECRH, ICRH) and fuelling and density control). In parallel, aspects of the HELIAS-optimization are prepared to be assessed as far as the heating capabilities and the achievable plasma beta will allow in the first phase. The prioritized physics topics are - in addition to the aforementioned subjects - 3D impurity transport, turbulence and its interplay with neoclassical transport and the preparation of fast particle studies for later phases. The European program is also to exploit synergies with the European contributions to ITER (e.g. ECRH) and DEMO and to continuously develop a physics basis for a stellarator fusion power plant.

The presentation reflects the results of the EUROfusion contributions and describes the scientific strategy of the EUROfusion exploitation of W7-X in view of the stellarator mission of the Roadmap.

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Experimental study of impurity transport in SOL region of HL-2A tokamak based on VUV spectroscopy and its numerical analysis with EMC3-EIRENE

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The roles played by impurities have been extensively studied in many magnetically confined fusion devices because the impurity strongly affects the power balance, the dilution of fuel ions and the mitigation of divertor heat load, etc. In ITER a reduction of impurity concentration in the core plasma is strictly required while a strong radiative cooling in the edge plasma is contrastively desired for the divertor heat flux mitigation. Therefore, the impurity transport study is of great importance not only for the plasma core but also for the plasma edge. Recent studies on both the experiment and numerical modelling in the tokamak configuration show that the edge impurity transport in the scrape-off layer (SOL), where the edge impurity transport is generally dominated by the parallel transport, is more sensitive to the source location compared to the stochastic magnetic field layer of the stellarator configuration, where the cross-field transport is also important in addition to the parallel transport. In the HL-2A tokamak with double-null divertor the impact of the source location on the edge impurity transport has been studied by changing the edge magnetic field. A clear change of the impurity profile in the scrape-off layer (SOL) has been observed for different impurity source locations, i.e. divertor target, dome plates and mid-plane limiter. A numerical simulation with three-dimensional edge plasma transport code, EMC3-EIRENE, suggests that a poloidal asymmetry in the impurity flow profile and an enhanced physical sputtering play an important role for the edge impurity distribution, in particular, the screening efficiency of impurities in the SOL region of HL-2A tokamak.

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Helium screening by the edge stochastic layer at the X-point in LHD

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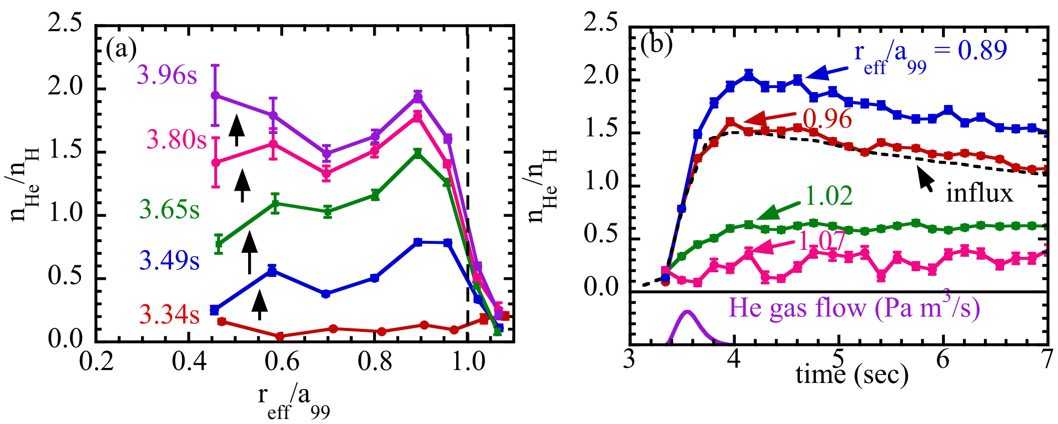
Radial profiles of density ratio of helium to hydrogen ions are measured using charge exchange spectroscopy technique with two-wavelength spectrometer in the Large Helical Device. Figure 1 shows the radial profiles and time evolution of density ratio of helium to hydrogen in the discharge with helium puff. The density ratio of helium to hydrogen increases after the He gas puff and reaches to 2.0. It should be noted that the $n_{\rm He}/n_{\rm H}$ value outside LCFS stays at a low level of 0.3. This result shows that the helium density outside LCFS (X-point region) does not increase even if helium gas is puffed. The time evolution of $n_{\rm He}/n_{\rm H}$ and the ratio of influx are plotted in Fig. 1(b). The He gas puff is applied at t = 3.4 s and the gas flow exceeds 10 Pa m^3/s and decreases to zero at = 3.8 s. Because the recycling rate of helium is higher than that of hydrogen, the $n_{\rm He}/n_{\rm H}$ value continue to increase after the turning-off of the He gas puff and reaches the maximum value of 2.0 at t = 4.1 s. The $n_{\rm He}/n_{\rm H}$ value inside LCFS is higher than the influx ratio, while that outside the LCFS is lower than the influx ratio. This result clearly shows that the density ratio in the plasma is not identical to the influx ratio because of the difference in penetration and transport between helium and hydrogen. The decrease of $n_{\rm He}/n_{\rm H}$ value outside LCFS is not due to the low temperature. The electron density and temperature are high enough to make the helium become fully ionized. The electron density and temperature measured with YAG Thomson scattering at the LCFS is 1.5-4.0$\times 10^{19}$m^{-3} and 0.3-0.4 keV. The $n_{\rm He}/n_{\rm H}$ ratio is measured at the mid plane where the helical X-point and stochastic layer locates just outside the LCFS. These observations demonstrates the helium impurity screening by the edge stochastic layer near the X-point.



(a)Radial profiles and (b)time evolution of density ratio of helium to hydrogen in the discharge with helium puff.

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Observation of carbon impurity flows in the ergodic layer of LHD using a space-resolved 3 m normal incidence VUV spectrometer

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Effects of edge stochastic magnetic fields on the impurity transport have attracted attention widely. Reduction of the cross-field impurity transport, so called “impurity screening”, has been observed in Large Helical Device (LHD) which has a thick stochastic magnetic field layer called “ergodic layer” located outside the core plasma [1]. Theoretically it is said that a parallel impurity momentum balance in the ergodic layer determines direction and quantity of the impurity flow, which could be one of key mechanisms of the impurity screening. Therefore, a precise measurement on the spatial profile of impurity flows is required as an experimental approach to investigate the impurity transport in such stochastic magnetic field.

We measured a carbon impurity flow in the ergodic layer of LHD by a vacuum ultraviolet (VUV) spectroscopy. Line radiations from impurity ions are significantly emitted in the ergodic layer because the electron temperature around the last closed flux surface (LCFS) ranges from 10 to 500 eV. Therefore, space-resolved spectroscopy using a 3 m normal incidence VUV spectrometer was developed to measure the VUV emission profiles in wavelength range of 300-3200 Å from impurities in the ergodic layer [2]. The emission intensity, the ion temperature, the impurity ion flow, and their vertical profiles are derived by measuring the Doppler profile of impurity line spectra.

The carbon impurity flows were derived by the Doppler shift of the CIV spectra with the wavelength of 1548.20 × 2 Å. We observed a clear increase of the flow in the ergodic layer when the averaged electron density was increased from 0.8 to 5.5 × 10^{13} cm^{-3}. A three-dimensional simulation code EMC3-EIRINE predicts that a friction force between bulk ions and impurity ions becomes dominant in the parallel impurity momentum balance in the ergodic layer when the electron density increases [3]. It results in impurity flows toward the divertor plates causing the impurity screening phenomena. Comparison of results between experiment and calculation will be discussed for quantitative evaluation of the edge impurity flows.

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Impurities in a non-axisymmetric plasma: transport and effect on bootstrap current

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Impurities lead to radiation losses and plasma dilution, and in stellarator plasmas the neoclassical ambipolar radial electric field often causes impurity accumulation in the core. In this work we use a new continuum drift-kinetic solver, the SFINCS code (the Stellarator Fokker-Planck Iterative Neoclassical Conservative Solver) [1] which employs the full linearized Fokker-Planck operator, to calculate neoclassical impurity transport coefficients for a Wendelstein 7-X magnetic configuration. The SFINCS calculations are compared with theoretical asymptotes in the high collisionality limit [2]. We find that in the low collisionality regime pitch-angle scattering is sufficient to accurately describe impurity transport through self-interaction, but lacks the ability to describe inter-species interaction. Applying a momentum correction technique to the pitch-angle scattering results is often, but not always, sufficient to reproduce the inter-species results obtained with the full linearized Fokker-Planck operator. At high collisionality, a momentum conserving collision operator is critical to correctly determine the impurity transport coefficients and a simple pitch-angle scattering approximation can lead to transport predictions in the wrong direction. We also use SFINCS to analyze how the impurity content affects the neoclassical impurity dynamics and the bootstrap current. We find that a change in the plasma effective charge Z_{eff} of order unity can affect the bootstrap current enough to cause a deviation in the divertor strike point locations.

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Comparative Study of Pellet Fuelling in 3-D Magnetically Confined Plasma Devices

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Core plasma fuelling is a critical issue on the path to developing steady-state scenarios in 3-D magnetically confined devices. Moreover, a detailed understanding of pellet ablation mechanisms, and subsequent particle transport to mitigate potential core particle depletion, is of high interest for the HELIAS line. In this paper, pellet injection studies performed on the LHD and TJ-II stellarator devices, with relevant parameter variations that complement comparative studies, are described, analysed and discussed. Indeed, the results presented here will provide a valuable input for the W7-X stellarator pellet injection program which is of critical importance for neoclassical transport optimization in W7-X and the attainment of long-pulse high-power discharges with pure microwave heating (the "standard" W7-X discharge scenario) with central particle fuelling attained by pellets.

In previous experiments made on LHD to study the reversal of diffusive particle fluxes due to hollow density profiles, the findings did not indicate anticipated effects in the radial electric field, rather they showed that, for appropriate pellet sequences, peripherally ablated pellets may lead to a mitigation of central density depletion. More recently, new experiments have been performed on LHD to elucidate on these findings while additional experiments have been made on TJ-II in order to perform a first comparative study. For this, sequences of pellets were injected in bunches of 1 to 4 pellets ($\sim 10^{21}$ particles/pellet) into LHD for different heating and target density regimes, *i.e.* counter and/or co-NBI with or without ECRH, in order o study the coupling of heat and particle sources and pump-out effects under various magnetic configurations so as to assess ripple effects on transport and effects of magnetic well/hill on the ablation process (grad B), target density and temperature. In parallel, up to 3 pellets (with ~1 to $\sim 4 \times 10^{19}$ particles/pellet) were injected into TJ-II plasmas maintained using NBI heating for a range of target densities and heating configurations. Both devices are equipped with a broad range of diagnostics that permit the pellet ablation to be followed and localized and that allow the subsequent particle drift and deposition to be evaluated and quantified.

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Perturbative particle transport experiments with pellet injection

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Core density control is a critical issue on the path towards the development of steady-state scenarios in 3-D magnetically confined devices, specially for the HELIAS line [1]. Therefore, an accurate and precise estimate of core particle transport is of critical interest in order to assess the risk of potential core depletion in large stellarators. This interest has triggered a set of inter-machine comparative pellet fuelling studies [2] whose goal is to achieve a detailed understanding of pellet ablation mechanisms and subsequent particle transport. In order to contribute to this research line, a compact pellet injector has been recently commisioned for the TJ-II stellarator; the system, associated diagnostics and first injections have been reported elsewhere [3].

In this contribution, we report particle transport studies in experiments with injection of small pellets in the stellarator TJ-II. The goal of these studies is twofold. On the one hand we aim at accurately characterizing particle transport as part of an inter-machine model validation activity [4]. Standard calculations (e.g. with ASTRA or TASK-3D transport suites) consist of calculating the particle source and then extracting the experimental particle flux from the particle balance equation. Nevertheless, this kind of calculation has the drawback of an incomplete knowledge of the particle source: its precise calculation would require self-consistent modelling of plasma edge transport and plasma wall interaction, which is still beyond the state of the art. By means of perturbative experiments we try to circunvent this problem and estimate the experimental particle flux directly. This experimental flux is compared to balance calculations with ASTRA and to neoclassical simulations with DKES and FORTEC-3D codes. On the other hand, we perform similar experiments in which two consecutive small pellets are injected into ECH-heated plasmas. With these experiments we aim at qualifying core density control schemes.

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Evaluation of neoclassical impurity transport coefficients at W7-X

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At W7-X, the target-released C- and the wall-released Fe-atoms are the major intrinsic impurity species, which influence the confinement performance of W7-X in terms of radiation. Their concentration in the confinement core is associated to the source strength and to their transport behaviour as well. Concerning impurity transport in W7-X, one of the most interesting questions is to what extent it follows the neoclassical expectation. For this purpose, the 1D impurity transport code STRAHL [1] has been recently coupled with the NTSS code [2]. The former provide the temporal-spatial evolution of impurity concentrations assuming diffusive and convective transports with sources at the plasma edge; the latter, based on mono-energetic coefficients in the DKES-database [3,4], which are determined by solving the linearized drift kinetic equation using the DKES code [5], performs convolutions of the mono-energetic coefficients with a local Maxwellian to calculate the particle (thermal) transport coefficients and provide the background plasma profiles for STRAHL. The radial electric field is determined using the ambipolarity constraint assuming small impurity contributions (i.e. tracer limit). For an ECR-heated hydrogen plasma with a central electron temperature of 6 keV, intrinsic impurity ions in the plasma core are predominantly C6+ and Fe-ions with high ionization stages (Z=23-26). Evaluations have been carried out for these ions in the standard magnetic configuration of W7-X. The obtained diffusion coefficients decrease in the plasma periphery for all ions; in the inner region the maximum value for C6+ reaches 0.05 m^2/s and for Fe-ions around 0.01 m^2/s; negative convection velocity is obtained outside the central heating power deposition region corresponding to inward drift of the impurity ions. Convective velocities for C- and Fe-ions reach -1.5 m/s and -2.5 m/s, respectively, at a radial position close to that having the strongest negative electric field. The evident Z-dependence of the diffusion coefficient in the core demonstrates that the impurity ions lie in the $\sqrt{\upsilon}$ -regime where the diffusion coefficient scales as $\sim Z^{-1}$. Higher Z-ions shows larger inward convective velocity in the outer region, but with a weak directly proportional Z-dependence. The calculations are performed keeping Z_{eff} (0)<1.05. The first results will be presented.

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ECE diagnostic for Wendelstein 7-X

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Among the set of diagnostics considered as mandatory for the test operation of W7-X is the Electron Cyclotron Emission Diagnostic, ECE. In first order the system is dedicated to track the plasma evolution and measure the electron temperature profile, the latter provided that the plasma optical thickness is sufficient for an interpretation of the emission as blackbody radiation. Moreover, the ECRH power deposition profile and dynamic electron heat transport could be derived from heatwave analysis modulating the power of a gyrotron.

The ECE diagnostic measures the 2nd harmonic x-mode emission at 2.5 T operation by means of a 32 channel heterodyne radiometer in the frequency band 126 GHz to 162 GHz. The toroidal position has been selected such that at the plasma axis differs from the 2.5 T in the ECRH launching plane, thus the central plasma temperature can be measured without being masked by the strong 140 GHz microwave stray radiation from the heating beams. Spatial resolution is maximized by a slim Gaussian beam probing perpendicular to the flux surfaces. The transmission line comprises a broadband horn with fundamental mode exit defining the polarization already in the vessel, a Viton-sealed mica sheet as broadband vacuum window and subsequent 28 mm diameter oversized circular waveguides with overall length ~22m to the radiometer outside the experiment hall. In front of the radiometer the frequency band of stray radiation from the various gyrotrons (139.9 to 140.4 GHz) is cut out of the spectrum by a waveguide Bragg reflection notch filter. The radiometer uses a single broadband mixer for down conversion and subsequent 2-18 GHz and 18-40 GHz filterbanks, 16 channels each. The bandwidth of the individual filters (0.25-1.4 GHz) corresponding to a radial resolution between 0.5 to 1cm is adapted to the radial resolution of the ECE emission, as calculated from the optical depth at the expected plasma conditions. The detector diode output is launched and filtered via differential amplifiers to the DAQ with maximum 2 Ms/s which allows to study fast mesoscale events also. For higher spectral resolution a zoom IF-device is available in parallel to the standard filterbank which allows the selection of any suitable frequency range of the spectrum by the aid of a tunable second local oscillator. In magnetic radial coordinates the selected frequency span of 4 GHz covered with 16 channels corresponds to a radial range of Δr~6cm at the High-Field Side (HFS) or ~15cm at the Low Field Side (LFS), respectively. For an overall absolute calibration of the diagnostic a second identical Gaussian optical frontend is installed as a twin outside the torus including identical waveguide components, mica window and a geometrically identical transmission line, however with a hot-cold calibration source chopping between LN_2 temperature and room temperature in front of it.

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Endoscope diagnostic for tomography, spectroscopy and thermography on Wendelstein 7-X

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Plasma–surface interaction (PSI) in the divertor region of Wendelstein 7-X (W7-X) will be of great interest and importance for the success of the operational phase OP1.2. While the erosion of divertor tiles will have an impact on the divertor's lifetime and is therefore a critical subject of investigation, fundamental PSI studies in the divertor region are in many ways equally significant.

These plasma–wall interactions will be influenced by impurity transport, where the complex three-dimensional magnetic geometry of W7-X will play a crucial role, but this magnetic geometry itself could also be influenced by plasma effects such as Pfirsch–Schlüter and bootstrap currents. Therefore, in addition to measurements of obvious quantities such as the heat flux onto the divertor tiles, research on plasma–wall interactions in the divertor region will require localized measurements of the temperature in the plasma edge and of the concentration and distribution of different impurities, in combination with modelling using impurity transport codes.

In order to provide the measurements necessary to address these physics questions, a set of endoscopes has been designed for visible and ultraviolet spectroscopy and tomography of the plasma edge, along with infrared thermography of the divertor tiles. The design, partly based on an endoscope system operating successfully on the Joint European Torus (JET), consists of four endoscopes arranged in pairs to aid the tomographic reconstruction. The endoscopes will employ entirely mirror-based optics in order to accommodate the wide range of wavelengths required for the diverse measurements and will have a rotating mirror in order to increase the field of view without compromising spatial resolution.

An overview of this endoscope diagnostic system will be presented. Details of the measurements to be taken and their relationship to physics issues such as impurity transport and erosion of the divertor will be discussed, with a particular emphasis on tomographic reconstruction algorithms.

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Prospects of X-ray Imaging Spectrometer for Impurity Diagnostic at W7-X

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For measurements of electron and ion temperature profiles (T_e ,T_i), poloidal plasma rotation (Φ_{rot}), and impurity density profiles (n_Z), two x-ray imaging crystal spectrometer systems are prepared for the first operational phase (OP1.1) of the advanced stellarator Wendelstein 7-X (W7-X) [1]. Both spectrometers use spherically bent crystals to create a 2D image on a CCD detector with spatial and energy resolution along vertical and horizontal direction, respectively. The spectrometer lines of sight are located in two different poloidal plasma cross sections, see Fig.1.

While the x-ray imaging crystal spectrometer (XICS) [2] has been designed for use with two crystals for a simultaneous measurement of spectral emission of Ar and Fe impurities in different charge states within a wide temperature range (T_e = 0.3 - 6 keV), the high resolution x-ray imaging spectrometer (HR-XIS) [3,4] can be equipped with up to 8 different crystals, mounted on a rotary stage to measure the emission of several intrinsic or injected impurities, depending on the crystal choice. Thus, in combination with *e.g.* a pulsed Ar injection, impurity pellet or a laser blow off system, the HR-XIS can be used for dedicated impurity transport studies at W7-X.

The design, capabilities, and possible future upgrades of both diagnostics will be discussed in detail.

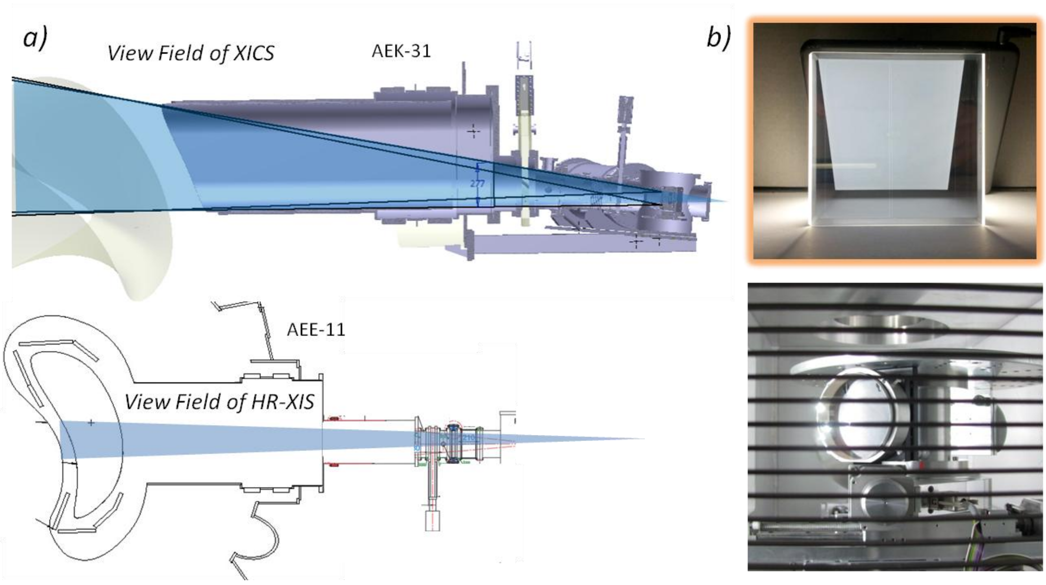

Fig.1: a) Viewing geometries and b) crystal designs of the XICS and HR-XIS diagnostics.

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Soft X-ray diagnostics for W7-X designed by IPPLM

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The investigation of X-ray spectra from plasmas is a standard diagnostic tool in fusion experiments. Measurements of X–ray intensities are usually realized by the use of Si detectors, which are sensitive to total radiation above a threshold energy determined by thin absorber foils placed in front of the diodes. They yield an excellent spatial and temporal resolution.

Two spectroscopic systems, pulse height analysis (PHA) and multi-filter (MFS), were designed for Wendelstein 7-X stellarator by IPPLM in cooperation with IPP.

The PHA diagnostic system, which is ready for mounting on W7-X, is intended to provide X-ray energy distributions at energy resolution not worse than 180 eV (for ^{55}Fe). The system consists of three spectrometry channels ranging from 350 eV to 20 keV. It is equipped with three Silicon Drift Detectors (SDDs) filtered with different Be filters and low noise charge preamplifiers. The detectors operate in a counting mode. The system will be installed on the horizontal port AEK50 of W7-X.

The MFS system is a multi-detector system designed for recording both, space and energy distributions of X-ray emission from plasmas. To recover energy distribution, the filter method is used. This diagnostic will consist of 40 silicon semiconductor detectors distributed as an 8×5 matrix. The detectors are accompanied with properly placed pinholes and Be filters. The detectors operate in a current mode and deliver continuous signal, which is proportional to the intensity of X-ray emission. The MFS system will be installed on the port AEN20 as a part of the so-called flexible SX camera system on the AEM10-AEN10-AEO10 port combination at W7-X.

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Assessing short-wavelength Alfvén resonance heating in H-1 heliac

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Alfvén resonance heating is based on the wave conversion phenomenon that occurs at the surface of the Alfvén resonance. The fast wave electromagnetic field is excited in plasma by the antenna and, converts to the slow wave. The slow wave, the quasi-electrostatic slow wave in cold plasma and the kinetic Alfvén wave in hot plasma, is strongly damped owing to the Landau mechanism and transfers its energy to plasma electrons. The short wavelength Alfvén resonance heating can be performed using sufficiently compact antennas, and the three-half turn antenna (THTA) is one of them. It has three straps oriented perpendicularly to the steady magnetic field. The side straps are fed with half-valued π-phased currents. The THTA is successfully used at Uragan-3M torsatron in Alfvén resonance heating regime [1].

The Alfvén resonance heating at H-1 with THTA is analyzed numerically using a 1D cylindrical full-wave code. The code accounts finite electron and ion temperatures. The power deposition profile and antenna loading resistance are analyzed depending on plasma density, temperature and heating frequency. The optimum antenna size is chosen. The important factor in the study is that the version of THTA for H-1 should follow the plasma column helix, thus the straps should be turned around the magnetic axis. The calculations show that this turning, as well as the magnetic field radial non-uniformity, results in some non-dramatic worsening of the power deposition profile. The antenna Q could be achieved below 40 when the frequency is 0.6-0.7 of the ion cyclotron frequency. The heating is efficient in sufficiently broad range of the discharge parameters.

The antenna specific design for H-1 heliac is presented and discussed.

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Neutral particle source and particle balance in the HSX edge

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This work is part of a larger effort to understand neutral, plasma source, and particle balance physics in the HSX stellarator. The HSX electron density is limited due to the ECRH cutoff density ($n_{e,cutoff} = 1x10^{19}$ m^{-3}), so neutral particles are predicted to have an ionization length on the order of the HSX minor radius (15 cm) or longer. This means that neutrals and plasma source are likely present throughout the HSX plasma, so it is important to understand the implications of this deeply penetrating neutral population.

As part of a larger effort to directly measure neutral profiles in HSX, we present preliminary measurements of neutral and singly ionized helium profiles, as well as the ionization length determined from the neutral helium density profile. Spectroscopic measurements of a neutral helium line and singly ionized helium line, together with ADAS data [1], are used to calculate the helium profiles.

Additionally, we present the measured gas puff neutral flux. For each plasma discharge, this quantity is determined using a previously obtained calibration matrix. We obtain the gas puff and wall recycling fuelling efficiencies numerically from both DEGAS [2] and EMC3-EIRENE [3]. These are essential for determining particle balance in HSX with a single reservoir model [4].

Finally, we have built and installed a movable limiter that can be positioned at the LCFS in order to concentrate particle and heat flux at a known location. We present the first unfiltered CCD images of the heat flux footprint on the limiter. These images suggest that the region of maximum thermal load agrees qualitatively with EMC3-EIRENE predictions. The results are necessary for estimating particle diffusivity in the HSX edge.

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On the neutral behavior under detachment conditions in W7-AS and W7-X

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In the W7-AS stellarator the subdivertor pressures behaved very differently on top and at bottom of the device [1]. The highest neutral pressure was observed under detachment conditions where the neutral pressure measured in the up- and down divertor chamber was usually strongly asymmetric. After detachment transition the neutral pressure in one divertor chamber increases almostly by a factor of 3, while less changed or even slightly decreased in the other, depending on the B-field direction. This asymmetry in neutral pressure cannot be produced by EMC3-EIRENE modeling so far due to the absence of drift terms in the model.

While the possibly drift-driven asymmetries still remain an open issue, we re-examine the neutral behaviour of the detached plasmas at W7-AS using the EMC3-EIRENE code without volume recombination and assess the role of volume combination in the neutral pressure rise based on experimental measurements [2], which is strongly indicated by spectroscopic diagnostics.

The high neutral density observed in a limiter case at TEXTOR has been explained as a consequence of a hydrogen Marfe [3]. We will examine the relevance of the TOKAR-model to the W7-AS divertor conditions to assess whether a similar effect can occur in W7-AS. A hydrogen Marfe is not a thermal instability like the impurity Marfe discovered in ALCATOR C-mod but an instability of the plasma-wall transition (ionization instability).

Finally, we present and discuss a prediction of the behaviour of the neutral pressure in the W7-X stellarator using the EMC3-Eirene model.

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Stellarators sources of ions for accelerators - symplectic calculations of ion losses

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Typical Electron Cyclotron Resonance Ion Sources (ECRIS) for accelerators such as the Large Hadron Collider are magnetic mirrors with a super-imposed hexapole field, confining plasmas of lead, gold or other elements [1]. ECRIS progressed to higher and higher ion currents and charge-states by adopting stronger magnetic fields (beneficial for confinement) and proportionally higher ECR frequencies. Further improvements would require the attainment of "triple products" of density, temperature and confinement time comparable with major fusion experiments. It is thus proposed that a toroidal rather than linear ECRIS would at the same time reach higher confinement, higher triple product, and make better use of the magnetic field, in the sense that, unlike a mirror, it would not need to generate a maximum field much higher than the EC-resonant field. The simplest toroidal generalization of a linear ECRIS is a classical stellarator of l=3. Ion extraction is more complicated than from a linear ECRIS but feasible, for example by means of magnetic deflectors or $\mathbf{E}\times\mathbf{B}$ extractors, as suggested by recent single-particle tracings [2]. New single-particle trajectories are presented in this work, which were obtained with a symplectic integrator. The faster integration allowed to model a larger number of particles and so predict regions where ion losses will be more significant and extractors would be more effective. Additionally, a random walk was super-imposed to simulate collisions in toroidal geometry.

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Evolution of radiation structure by three dimensional measurement during radiation collapse in LHD

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When the density of the plasma is increased, the fusion output and the energy confinement time of a confined plasma are increased. For the achievement of future fusion devices, high density operation is preferred. Helical plasmas such as Large Helical Device (LHD) plasma are suitable for high density operation because the helical plasma is free from the possibility of disruptions. However the density of the helical plasma is limited by a density limit [1]. When the density approaches the density limit, the plasma is terminated by radiation collapse. Therefore, an understanding of radiation collapse is important for the operation of future fusion devices. In helical devices, the radiation has a three dimensional structure (3D). To measure the radiation, a 3D radiation measurement had been developed using a technique of 3D tomography in the LHD [2] [3]. In this study, the radiation collapse in LHD plasma has been investigated using four infrared imaging video bolometers (IRVBs) [4] and the 3D radiation measurement thereby.

In LHD, IRVBs provide 2D radiation images with a 40 ms time resolution. The evolution of radiation structure during radiation collapse with density ramp up was measured by the IRVBs and then the technique of 3D tomography was carried out using these images. The result of the 3D radiation measurement show that an enhancement of the local radiation from the inboard side initiates at the vertically elongated cross section and then it is extended to the other poloidal cross sections along the last closed flux surface. The inboard side of the vertically elongated cross section is at the nearest point to the wall in LHD. The result also shows that radiation region minor radially localized before radiation collapse. In this phase, the mean free path suddenly drops nonlinearly with increasing electron density. This behaviour indicates that the structural change during radiation collapse is related to the reduction in parallel transport.

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Session 5

Reactor perspectives

FROM W7-X TO HELIAS: STRATEGY AND PROGRESS TOWARDS A STELLARATOR POWER PLANT

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With the completion of W7-X and the start of its exploitation to demonstrate the reactor capability of the stellarator line renewed focus is set upon the vision of a helical-axis advanced stellarator (HELIAS) power plant. Correspondingly the question is raised concerning the properties of such a device, how they compare with tokamak designs and which steps must be taken to arrive at a conceptual HELIAS design including the identification and preparation of dedicated experiments needed in W7-X.

In order to investigate these questions a systematic strategy is followed: Starting from a systems code – being a comprehensive model of an entire fusion power plant – systems studies are carried out in order to identify the accessible design window of a HELIAS power plant. These studies over a wide range of parameters are then complemented by detailed physics and technology simulations, i.e. a combination of neoclassical and turbulent transport simulations, as well as verification in W7-X experiments.

As the newly developed stellarator-specific systems code modules were implemented in the widely used tokamak framework PROCESS, a representative conceptual design point can be selected and compared against an equivalent tokamak under similar assumptions and goals. It turns out that possible HELIAS design points exist even under conservative assumptions and that the total construction costs are similar to these of a tokamak.

Since the step from W7-X to such a HELIAS power plant would be very large both in engineering and physics quantities, a risk-reducing strategy foresees an intermediate step stellarator to bridge this gap. Several different concepts can be investigated for such a device ranging from a fast-track, cost-efficient device with low field and without blanket to a nearly DEMO-like full engineering machine. The costs from the smallest reasonable machine to a DEMO-like device would increase roughly by a factor two while additionally requiring considerable technological development.

To qualify the engineering results from the systems code approach collaboration with KIT has been initiated with the aim of a nuclear analysis and blanket optimization of HELIAS devices. The 3D geometry of the stellarator poses a considerable challenge in this activity as the well-known MCNP code can only handle surfaces with a maximum quadratic description which is in contrast to the splines used in the stellarator CAD models.

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Examination of plasma operation control scenario and operation regime of the LHD-type helical reactor FFHR

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Plasma operation control scenario of the LHD-type helical reactor FFHR-d1 was examined using the integrated 1D physics model based on the LHD experimental observations [1]. It was found that a smooth change of the fusion power and steady-state sustainment of self-ignition operation condition can be achieved by a relatively simple control method with a small number of simple diagnostics: feedback control of the pellet fuelling by the measurement of the line averaged electron density and a staged variation of the external heating power based on the measurement of the edge density and the fusion power. Consistency with MHD equilibrium and neo-classical transport was confirmed by coupling with the integrated transport analysis code TASK3D. This operation control method is insensitive to the plasma properties (e.g. helium ash fraction, alpha heating efficiency, confinement improvement factor, etc.) as long as the target line-averaged electron density and fusion power are properly set. However, the possible operation regime strongly depends on the plasma properties. In the case of self-ignition operation, achievable minimum value of the fusion power increases with increasing helium ash fraction or decreasing alpha heating efficiency. Consequently the design regime is limited due to the increase of the heat and particle load on the plasma facing components. In the case of sub-ignition operation, the required heating power increases and the design regime is limited due to the decrease of the engineering Q value. Increase of the magnetic field is an effective countermeasure to enlarge the operation regime, but it will be limited by the engineering aspects.

In the conceptual design activity of the helical reactor FFHR, a multi-path design strategy has been introduced [2]. To strengthen the design feasibility, it considers a flexible selection of the reactor size and the magnetic field strength, including sub-ignition operation. In the presentation, a surely reachable design and operation regime based on the extrapolation of the present LHD experimental results is shown and physics issues which strongly related to the enlargement of the design and operation regime is discussed.

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Plasma disruption avoidance using non-axisymmetric shaping with stellarator fields

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The avoidance and mitigation of major disruptions remains a critical challenge for ITER and future burning plasmas based upon the tokamak. The addition of small 3D magnetic field perturbations has been used for a variety of beneficial purposes on present day tokamak plasmas [1]. Early stellarator experiments with toroidal plasma current were found to operate without disruptions if the vacuum rotational transform produced by external coils was greater than a threshold value of $\iota_{vac}(a) > 0.14$ [2]. Strong 3-D shaping produced by externally generated rotational transform is also observed to suppress disruptive phenomena of current-carrying discharges in the Compact Toroidal Hybrid (CTH), with the amount of rotational transform, $\iota_{vac}(a)$, required for suppression dependent upon the disruption scenario. Current-driven disruptions are deliberately generated in CTH by (1) raising the plasma density, (2) operating at low edge safety factor $q(a)$, or (3) by not compensating against the vertical instability of plasmas with high elongation. While the density limit is found to agree with the empirical Greenwald limit at low edge vacuum transform $\iota_{vac}(a) = 0.04$ the experimental densities exceed this limit by up to a factor of three as the vacuum transform is raised to $\iota_{vac}(a) = 0.25$ without a threshold value of $\iota_{vac}(a)$ observed for elimination of disruptions. Low-q disruptions near $q(a) = 2$ are observed at low vacuum transform but appear to have a threshold transform for avoidance, no longer occurring when the vacuum transform is raised above $\iota_{vac}(a) > 0.07$ even though $q(a)$ falls well below a value of 2. Passive suppression of the vertical instability of elongated plasmas is also observed with the addition of external transform [3], and the amount required is in qualitative agreement with an analytic calculation of marginal stability in current-carrying stellarators [4].

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ICRH as a Dedicated Fast-Ion Source for W7-X

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One of the main scientific objectives of W7-X is to demonstrate good confinement of energetic ions in this device [1]. This is a main requirement to extrapolate W7-X results towards a stellarator fusion reactor. Particles with energies of about 50–100 keV need to be generated in W7-X to mimic fusion-born alpha particles in a future HELIAS stellarator-reactor [2].

The magnetic configuration of W7-X is designed to confine high-energy ions close to the center. Good particle confinement requires high plasma beta $<\beta> \approx 4\%$, which is reached at very large plasma densities, $n_{e0} \simeq 2\times10^{20}\ \mathrm{m}^{-3}$. An ion cyclotron resonance heating (ICRH) system is planned to be installed in W7-X to produce such energetic ions under these operational conditions [3]. The ongoing ICRH system design for W7-X will use the TEXTOR generators, which should couple between 1 and 2 MW of ICRH power to the plasma within the frequency range $f = 25 – 38$ MHz.

In this paper, we report on the current status of the design of the ICRH system in W7-X. The emphasis is on using ICRH as a localized source of fast ions. Central ion heating scenarios relevant for hydrogen and deuterium phases of W7-X operation are outlined. We also discuss a relevance of the novel three-ion ICRF scenarios for fast-ion generation in W7-X [4]: in H:D ≈ 70%:30% plasmas, an efficient absorption of ICRF power by a very small fraction of helium-3 ions (~ 0.1%) is possible. This ICRF scenario has a potential to generate fast ^{3}He ions at the largest plasma density envisaged for operation in W7-X.

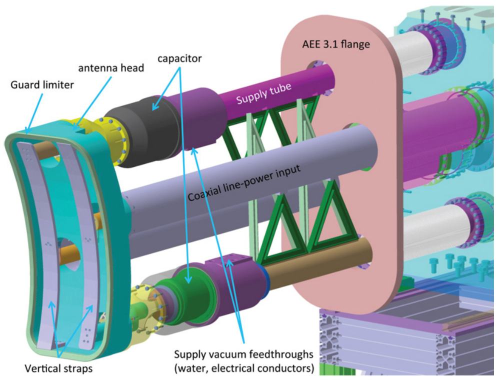

Figure 1. Sketch of the ICRH antenna system in W7-X.

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Bayesian model for temperature and density evaluation from W7-X Thomson scattering.

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An optimized helical axis stellarator is a promising path toward a fusion reactor. Wendelstein 7-X (W7-X) is the first fully optimized stellarator in this line [1]. Achievement of optimization criteria, which are high equilibrium beta, low Shafranov shift and bootstrap current, MHD stability, neoclassical transport, fast ion confinement, has yet to be demonstrated. For such verifications, Bayesian analysis is suitable, for which the Minerva framework [2] is employed at W7-X. This allows proper handling of uncertainties, and combination of information from different diagnostics. In this contribution a W7-X Thomson scattering diagnostic for electron temperature and density, a corresponding Minerva model and first measurements are presented.

The core W7-X Thomson scattering diagnostic is based on Nd:YAG lasers with 1 J per pulse and 20 Hz pulse repetition rate. Up to 5 lasers in a 2D arrangement [3] are foreseen, with 2 lasers being available in the first phase. Two front lenses with $\delta\Omega$ of about $1.2\cdot10^{-2}$ and $8\cdot10^{-3}$ sr observe the full plasma diameter. Scattered light at the lens image planes is focused onto fiber bundles and is guided to a set of polychromators. Initially 10 spatial locations are observed; later the system will be expanded to 60 channels. Fibers for two spatial channels are sent to a single polychromator, thereby saving the equipment. One of the fiber bundles in each pair is 20 m longer to provide temporal pulse separation. Each polyhcromator consists of 5 spectral channels, where filters are chosen for electron temperatures up to 10 keV. Light in each spectral channel is detected and amplified with an avalanche photodiode and sampled at 1 GSample/s using a fast ADC. Position calibration of the system is performed by using a laser tracking system with precision below 1 mm. Spectral calibration is done with the help of a supercontinuum white light laser. Absolute calibration is derived from Raman scattering in nitrogen.

To reconstruct electron temperature and density a Minerva (Bayesian) model is implemented. Given temperature and density the model predicts the recorded signals. The Minerva framework inverts the model and finds the most likely plasma parameters and their uncertainties. A systematic scan with this model predicts errors of about 5% for T_e and about 2% for n_e for the W7-X setup and T_e from 20 eV to 10 keV. The model was verified on test data from ASDEX Upgrade. First application to W7-X data is shown.

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Construction, Commissioning and first Results of the Costa Rican Stellarator SCR-1

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SCR-1 is a small-size stellarator of the modular-type designed, constructed and currently being implemented in Costa Rica (R_o=0.238 m, <a>=0.059 m, R_o/a>4.4, expected plasma volume ≈ 0.016 m^3, 10 mm thickness 6061-T6 aluminum vacuum vessel) [1,2,3]. The project is in its final assembly phase and is expected to start operation in june 2015. This contribution provides an overview of the SCR-1 project from the engineering and manufacturing point of view with particular emphasis on the comprehensive test program and its results. The magnetic field strength at the center is around 43.8 mT which will be produced by 12 copper modular coils with 4.6 kA-turn each. This field is EC resonant at R_o with a 2.45 GHz as 2nd harmonic, from 2 kW and 3kW magnetrons. SCR-1 was redesigned from stellarator UST_1 [4]. As a first step, the objectives focus on training human resources and identifying of problems related to the design and construction of small modular stellarators. The vacuum vessel was built using CNC machines in two aluminum blocks. The ports were specially designed for vacuum conditions. The first vacuum vessel test indicates a pressure of 10-7 Torr. All welds were made by TIG or GTAW technique. The coils were built using 3D printing, and casting molds. Simulations and calculations of magnetic field, magnetic surfaces, tracking particles, microwave transmission, heating wire-coils, mechanical rigidity of the vacuum vessel, and others were performed. The power supply consists of an industrial batteries bank (150 A-h, 120 V). An electric current regulator was designed to maintain constant current despite the heat in the coils, a 10 s pulse is expected. The diagnostics consist of a Langmuir Probe (two removable heads, four tips each one), an iHR550 optical spectrometer and a Heterodyne Microwave Interferometer (28 GHz). It has a module NI PXIe-8135 2.3 GHz Core i7-3610QE Controller, Win7 (32-bit) for data acquisition and control. Finally we provide our recommendations for constructing the small modular stellarators.

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Progress in development of stellarator-mirror fission-fusion hybrid concept

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Recent developments on a stellarator-mirror fission-fusion hybrid concept [1] are reviewed. The hybrid consists of a fusion neutron source and a powerful sub-critical fast fission reactor core. The aim is transmutation of spent nuclear fuel and safe fission energy production. In its fusion part, a stellarator-type system with an embedded magnetic mirror is used. The stellarator confines deuterium plasma with moderate temperature, 1-2 keV. In the magnetic mirror, a hot component of sloshing tritium ions is trapped. There the fusion neutrons are generated.

A candidate for a combined stellarator-mirror system is the DRACON magnetic trap. For high energy ions of tritium with an energy of 70 keV, comparative computations of collisionless losses in the mirror part of a specific design of the DRAKON type trap are carried out. Two versions of the trap are considered with different lengths of the rectilinear sections. Also the total number of current-carrying rings in the magnetic system is varied. The results predict that high energy ions from neutral beam injection can be satisfactorily confined in the mirror part during 0.1-1 s [2].

The MCNPX Monte-Carlo code has been used to model neutron processes in a fission reactor core that use a fuel made of reprocessed spent nuclear fuel. Fission rates for transuranic elements are calculated for two variants of the reactor coolant [3].

The magnetic configuration of a stellarator with an embedded magnetic mirror is arranged in the Uragan-2M experimental device by switching off one toroidal coil. The motion of particles magnetically trapped in the embedded mirror is analyzed numerically with use of motional invariants. It is found that without a radial electric field the particle quickly drift out of the stellarator-mirror. A weak radial electric field which could be spontaneously created by ambipolar radial particle losses can make the drift trajectories closed, which substantially improve particle confinement. It is remarkable that the improvement acts both for positive and negative charges. [4].

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Stellarator Optimisation with ROSE

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As W7-X is going into operation, the need for designing a next step stellarator is gaining urgency. Further improvement of fast particle confinement needs to be achieved in combination with a small bootstrap current. Additional changes of design rationale may emerge as W7-X delivers scientific results.

The ROSE (ROSE Optimises Stellarator Equilibria) was written to carry out nonlinear optimisation of stellarator equilibria. In addition to the usual optimisation criteria, ROSE also optimises for simplicity of coil design as well as many criteria relating to fast particle confinement

Two optimised stellarators are presented:

1. an optimised W7X-type equilibrium with strongly improved effective ripple and a large mirror term. Coils have been designed assuming current densities achievable with high-T_C superconductors currently becoming available.

2. a quasi-axially symmetric stellarator intended as university experiment featuring small effective ripple, modest rotational transform and very simple coil construction.

   Non-superconducting coils have been designed to faithfully reproduce the small ripple and hence keep QA symmetry with good accuracy.

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Options for an Intermediate-Step burning-plasma Stellarator

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A number of important scientific questions still need to be solved in order to achieve the dream of a sustainable electricity supply by fusion power plants in the future. One of these aspects is the confinement and behaviour of a burning plasma with considerable production of fusion power over a sufficient length of time. For the tokamak concept this question will be addressed by the well known international burning plasma experiment ITER. For the helical-axis advanced stellarator (HELIAS) line the reactor capability will be tested in the optimised Wendelstein 7-X experiment but investigation of stellarator burning plasma properties would need to be done in a machine beyond W7-X. As the 3D topology of the HELIAS introduces new physics aspects which cannot be treated by an axissymmetric machine like ITER, a dedicated HELIAS device may be required in order to mitigate the development risks leading to a stellarator power plant. Such a device would bridge the gap between current experiments and envisaged HELIAS power plants and is referred to here as an 'Intermediate Step Stellarator'.

Based on this motivation different concepts for an intermediate step stellarator are proposed based on different levels of dedication reaching from a small, near-term-ready machine without blanket up to a nearly DEMO-like concept including all essential technologies.

For each concept idea a design analysis has been carried out using the systems code PROCESS to define the possibilities for the realisation of each machine. The individual design points are compared in this common framework showing a factor of two differences in costs between the smallest and the largest of the concepts. Further criteria must be defined to make a sensible choice on which design shall be followed.

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Session 6

Coupling of core optimization to the plasma boundary and PMI

Setup and Initial Results from the Magnetic Flux Surface Diagnostics at Wendelstein 7-X

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In stellarators and heliotrons, the existence of closed and nested flux surfaces can be determined without plasma present. A flux surface diagnostic has been installed and tested in W7-X, which is currently under commissioning [1]. The diagnostic is based on the fluorescent technique utilizing a low energy electron beam that follows the magnetic field lines within the confinement region until it is intercepted by a fluorescent detector in a fixed plane, thereby creating a 2-dimensional Poincaré-plot of the magnetic flux surface [2, 3].

A manipulator has been built for a precise positioning and movement of electron emitters and detectors. Two identical manipulators have been installed in module 1 and 3; each one can be used either for moving the electron source or as the detector. By interchanging their functionality, measurements are possible at two different toroidal positions. The detection of the fluorescent signal is realized by sensitive cameras that are installed in a tangential ports looking onto the detection plane of the rods. As a reference system, and for scaling of the camera images, 4 externally illuminated metal coated optical fibers have been integrated within each detection plane. The position of the reference system and the orientation of the detection plane of the manipulator arm have been carefully measured within the plasma vessel by means of metrology.

Initial measurements are planned during the commissioning of the type “3” non-planar field coils which are already creating flux surfaces and for the magnetic configuration dedicated to the limiter operation in OP1.1 at magnetic field strength of 0.4T, 1.2T, 1.9T and 2.5T which allows checking structural properties of the global magnetic field system. Furthermore, measurements including the detection of possible B_{11} error fields leading to a shift of the magnetic axis for a resonant central iota of one, tests of the error field trim coils and the other magnetic standard configuration are foreseen as well. By comparing the flux surfaces from both toroidal positions first results on the stellarator symmetry can be concluded.

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Plasma material interaction with long time-scale between plasma edge and plasma facing component using steady-state plasmas in the LHD

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High performance (line averaged electron density: $n_e > 10^{19}$ m^{-3}, electron temperature at plasma center: $T_{e0} > 2$ keV, plasma heating power: $P_{RF} \sim$ 1MW) steady-state helium plasma with plasma duration time $t_d > 1000$ sec has been investigated using ion cyclotron heating (ICH) and electron cyclotron heating (ECH) in the Large Helical Device (LHD), and time evolution of wall-pumping effect associated with plasma material interaction with continuous helium plasma irradiation and erosion with heat flux (> a few MW/m^2) on divertor plates was clearly observed on the plasma duration time over 2000 sec. In lower performance helium plasma ($n_e \sim 0.4$x10^{19} m^{-3}, $T_{e0} < 1$ keV, $P_{RF} \sim$ 0.4MW, $t_d \sim$ 54 min. and $n_e \sim 0.1$x10^{19} m^{-3}, $T_{e0} < 1$ keV, $P_{RF} \sim$ 0.1MW, $t_d \sim$ 65 min.), wall-pumping was kept constant on long-pulse plasma duration, and high-performance and high-power steady-state helium plasma operation ($n_{e0} \sim 10^{19}$ m^{-3}, $T_{e0} \sim T_i \sim$ 2keV, $P_{RF} \sim$ 1MW, $t_d > 2000$ sec) was required to investigate plasma material interaction with long time-scale.

When high-performance long-pulse plasma were repeatedly maintained, spikes of carbon line emission, CIII, had been clearly observed with radio frequency heating power $P_{RF} > 1$ MW. Although contamination frequencies of carbon impurity were clearly increased, radiation power using a bolometer had been kept constant (~ 17%) in the plasma performance ($n_{e0} \sim 1.2$x10^{19} m^{-3}, $T_{e0} \sim T_i \sim$ 2keV, $P_{RF} \sim$ 1.2MW, $t_d \sim$ 2800 sec). Mainly impurity source came from carbon divertor plates with continuous erosion by physical sputtering, and carbon contamination was carried out by exfoliation of carbon rich deposition layer (mixed-material layer, MML) and explosion of helium bubbles in the MML. The MML with amorphous structure can retain the helium and hydrogen particles, and helium particles are easily released on the low temperature conditions (< 400 K).

In high-performance and high-power long-pulse plasma durations, the MML plays important role for particle retention and impurity contamination rather than short pulse plasma ($t_d < 10$ sec), and it will be key issue to control the growth and exfoliation of the deposition layer caused by continuous divertor heat flux in high-performance steady-state devices.

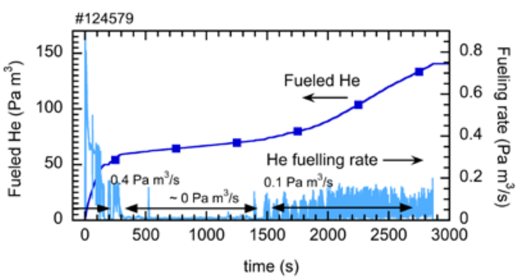

Fig.1 Time evolution of helium particle fueling with three phases.

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Numerical exploration of detachment performance on W7-X

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Divertor experience gathered from tokamaks and stellarators so far strongly indicate that, on the way to a reactor, a strategy must be developed to remove most of the thermal power from the SOL before reaching solid surfaces. For this purpose, radiation via light impurities is a concept that is intensively being explored across the world, both experimentally and theoretically. Thus, the reactor-relevance of a divertor concept is largely determined by its performance in a state of detachment characterized by enhanced radiation and reduced recycling. Here, the most critical issues are the maximum power removal capability of the divertor (including possible limiting factors), the particle-exhaust efficiency under degraded recycling conditions, impurity control under intensive radiation, and the stability of the detached state. Using the EMC3-Eirene code and modelling experience from W7-AS and LHD, this paper presents a numerical analysis of these issues for the W7-X island divertor, providing a reference for developing the experimental divertor programme on W7-X.

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Impact of 3D equilibrium response on edge topology and divertor heat loads in W7-X

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Wendelstein 7-X has been designed to minimze the effect of plasma currents so as to provide robust magnetic configurations well suited for quasi-steady-state operation with an island divertor concept. This requires an edge topology with boundary islands which define the interaction locations between the plasma and the divertor. However, the optimization of W7-X is a result of different requirements which forced a compromise with respect to different criteria. Thus, there are still plasma pressure effects on the equilibrium, and the bootstrap current minimization is restricted to a sub-space within the larger 6-dimensional space of magnetic configurations. In practice, the bootstrap current will produce a total net-toroidal current, which changes the magnetic boundary topology.

Up to now, these topological changes were studied mainly with the VMEC/EXTENDER code-combination. The virtual casing principle, which is used in the EXTENDER code, can give the magnetic field driven by the 3D equilibrium response calculated by VMEC. In studies with the VMEC/EXTENDER code, it has been found that net toroidal current significantly affects the magnetic boundary topology, mainly by a change in the boundary rotational transform leading to a radial displacment of the boundary islands crucial for divertor operation. This radial dislocation can be compensated by either external current drive (using ECCD, no ohmic transformer) or by properly adjusting the vacuum rotational transform. However, since VMEC is an inverse solver based on flux coordinates, the effect of the details of the perturbed pressure distribution on the magnetic boundary island cannot be considered. Therefore, the HINT-code, which is not limited by the assumption of nested flux surfaces or spectral resolution, has been applied to the magnetic boundary topology problem in W7-X for cases including net toroidal currents.

Using the HINT-code, the change of the magnetic topology connected with net toroidal currents are studied qualitatively and quantitatively. In addition, the effect of different boundary topologies on the divertor heat loads is studied as well as the changes of the deposition "footprints" on other in-vessel components.

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Overview of the U.S. Wendelstein 7-X Program*

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In 2010, a team of stellarator researchers from three U.S. National Laboratories launched a collaboration with the Max Planck Institute for Plasma Physics (IPP) to work in partnership, using the Wendelstein (W7-X) facility to advance the physics and technology of 3D magnetic confinement for fusion. A key U.S. research focus is to investigate plasma behaviour and control in 3D diverted stellarator. The first few years of the collaboration coincided with the last years of W7-X construction and commissioning. During that period, several U.S.-IPP joint projects have been carried out, including significant in-kind U.S. contributions to the W7-X facility. Under the collaboration, we have: 1) designed a procedure and specialized tooling that were successfully used to install the high-temperature superconducting current leads, 2) designed and implemented a complete trim coil system for controlling low-order field errors, 3) deployed a high-resolution camera for infrared imaging of in-vessel components, 4) designed a steady-state "scraper" to protect the divertor leading edges as well as an inertially-cooled prototype for testing during OP1.2, 5) fabricated an x-ray imaging crystal spectrometer for high-resolution temperature and velocity profile measurements starting from OP1.1, 6) developed tools for equilibrium reconstruction from experimental data including forward modelling of OP1.1 field mapping experiments, and 7) designed pellet mass detectors for use in pellet injection experiments starting with OP1.2.

With the start of W7-X commissioning in 2014, the U.S. team stepped up its preparations for participation in W7-X operation and physics research, making use of U.S.-furnished equipment and analysis tools. In addition, the U.S. team expanded with the addition of several new projects led by university and industry teams. Several of these new projects will support transport research, namely 1) design and deployment of a phase contrast imaging diagnostic, 2) design and initial equipment deployment in preparation for a gas puff imaging diagnostic, 3) conceptual design of a heavy ion beam probe diagnostic. A fourth new project will provide simulation studies and diagnostic tools for edge plasma equilibrium and stability and divertor studies.

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Current Status of the Hybrid Illinois Device for Research & Applications (HIDRA)

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TOPIC S6: Coupling of core optimization to the plasma boundary and PMI

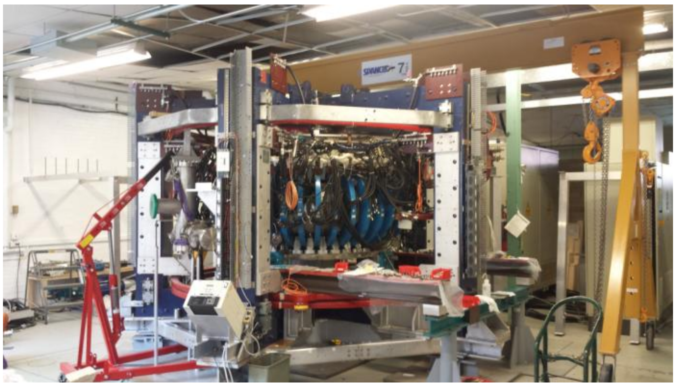

Figure 1: HIDRA being assembled at CPMI.

The Hybrid Illinois Device for Research and Applications (HIDRA) is the former WEGA stellarator (previously at MPIPP Greifswald) that is currently being re-assembled at the Center for Plasma Material Interactions (CPMI) at the University of Illinois, USA[1]. HIDRA is a medium sized classical stellarator using a l = 2, m = 5 configuration with R = 0.72 m, r = 0.19 m and a plasma radius a = 0.10 m. The magnetic field topology is set up by 40 toroidal field coils, 4 helical coils and 2 vertical field coils. Initial operation will have 26 kW of magnetron power (2.45 GHz) and with OXB heating is expected to reach densities of $n_e = 1\times10^{18}$ m^{-3} and temperatures T_e = 20 eV. The focus of HIDRA is plasma material interaction (PMI) research and plasma facing component (PFC) development. In fact HIDRA will be the first toroidal device dedicated to the development of new PMI experiments in a fusion type environments and multi–scale and multi–phase materials. The expertise of CPMI with *in-situ* diagnostics, such as MAPP[2], will open up new opportunities for innovative material testing[3,4] and play a leading role in the development of future PFC solutions for fusion, for example innovative liquid metal divertor designs such as LiMIT[5]. HIDRA will also play an important role in helping to validate computational modelling of edge plasmas and plasma surface interactions. Currently HIDRA is still being assembled with first vacuum expected in early July 2015 and the first plasma to be in September of 2015. This paper will present some of the initial measurements of HIDRA as it comes on line and will discuss some of the challenges encountered during assembly.

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Control of runaway electron flows in torsatrons Uragan-2M and Uragan-3M

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As known, appearance of runaway electrons (RE) is frequently observed in toroidal fusion devices: at the start of discharge in tokamaks, during additional plasma heating, or particles injection. As a rule, the presence of RE is parasitic phenomenon because of impact of high energy electrons on in-vessel components or the vessel walls that is resulting in their damage and is accompanied by intensive x-ray emission.

In this presentation we summarize the experimental and theoretical results obtained when investigating possibility to control behavior of RE flows in torsatrons Uragan-3M [1] and Uragan-2M [2], i.e., to suppress or, on the contrary, to stimulate their appearance.

The reason of electron acceleration to high energy is the unidirectional electric field caused by the time variation of currents in the helical winding at phases of current ramp-up and current ramp-down.

To suppress RE flows, a positive or negative potential was applied to the electrode located near the boundary of plasma confinement volume. To explain the mechanism of RE flow suppression a mathematical modeling was provided with calculations of motion of electrons in the presence of electrode being charged positively or negatively [1]. It was found that the results of calculations are in a quite good agreement with experimental results.

From the obtained results a quite common conclusion follows: the dynamic of runway electrons can be influenced by a spatial potential of plasma column produced either by supplying voltage to a special electrode or as a consequence of plasma heating. Therefore the method for suppression of RE flow occurrence, which we used for devices with pulsed magnetic field, can be of interest for cases when experiments are provided with stationary magnetic field but where some other reasons for appearance of RE flow occur.

In other experiments, the effect of RE flow stimulation was studied at both devices with an aim to stabilize the start of RF discharge. It was observed that reproducibility of discharge became much better.

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Plasma effects on separatrix-limited configurations of W7-X

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Wendelstein 7-X aims at quasi-steady-state operation to demonstrate the reactor-viability of stellarators optimized with respect to MHD-equilibrium and -stability, low neoclassical transport, small bootstrap current and good fast-particle confinement. To reach this goal an island divertor is foreseen for particle and energy exhaust, which utilizes the naturally occurring boundary islands connected with the appearance of low-order rational values of the rotational transform ι at the plasma boundary. The island separatrix thus bounds the plasma, and the strike lines of the island fans determine the heat load distribution on the divertor structures.

Although the configuration of W7-X has been optimized to display a small impact of plasma currents on the configuration, these effects still persist and change the plasma shape and the boundary islands' width and location. From previous studies it is known for example that with growing plasma-β the island width also increases [1, 2, 3], and the X- and O-point locations move poloidally, consistent with the effect of the Shafranov-shift. A net toroidal current is known to shift the island-generating resonance radially which, depending on the amount of plasma current, can lead to a detachment of the island structures from the divertor plates resulting in a limiter magnetic configuration [4].

For the present study the VMEC-EXTENDER code combination [3, 5] is utilized. The effect of plasma-β and of net-toroidal currents on the width and location of the islands is investigated in configurations in which the bootstrap current is expected to be small enough (according to transport simulations) to allow high-performance, quasi-steady-state operation [6] compatible with the island divertor. The change in magnetic configuration around the separatrix is compared with the accompanying changes in the spatial distribution of field lines with different connection lengths.

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