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Book of Abstracts

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	Monday Oct. 12, 09		Tuesday Oct. 13, 09		Wednesday Oct. 14, 09	Thursday Oct. 15, 09	Friday Oct. 16, 09
8:30		8:30	PL 01 Yamada	8:30	PL 02 Finken	PL 03 Hirsch	PL 04 Toi
8:45	opening	8:45		8:45			
9:00	I 01 Bosch	9:00		9:00			
9:15		9:15	I 09 Cooper	9:15	I 16 Jakubowski	I 23 Estrada	I 29 Spong
9:30	I 02 Blackwell	9:30		9:30			
9:45		9:45	I 10 Narushima	9:45	I 17 Unterberg	I 24 Sano	I 30 Isobe
10:00	I 03 Pankratov	10:00		10:00			
10:15		10:15	I 11 Pretty	10:15	I 18 Solomon	I 25 Xanthopoulos	I 31 Kobayashi
10:30	coffee	10:30		10:30			
10:45		10:45	coffee	10:45	coffee	coffee	coffee
11:00	I 04 Helander	11:00		11:00			
11:15		11:15	I 12 Reiman	11:15	I 19 Tabares	I 26 Ramisch	I 32 Murakami
11:30	I 05 Lore	11:30		11:30			
11:45		11:45	I 13 Hudson	11:45	I 20 Masuzaki	I 27 Pedrosa	I 33 Heidbrink
12:00	I 06 Ohdachi	12:00		12:00			
12:15		12:15	I 14 Hanson	12:15	I 21 Ding	I 28 Tamura	I 34 Suzuki
12:30	I 07 Yoshinuma	12:30		12:30			
12:45		12:45	lunch	12:45	lunch	lunch	closing
13:00	lunch	13:00		13:00			lunch
13:15		13:15		13:15			
13:30		13:30		13:30			
13:45		13:45	Poster (w. coffee)	13:45	Poster (w. coffee)	Excursion	
14:00	Poster (w. coffee)	14:00		14:00			
14:15		14:15		14:15			
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15:15		15:15		15:15			
15:30		15:30		15:30			
15:40		15:40	C 05 Canik	15:40	I 22 Masamune		
16:00	I 08 Morisaki	16:00	C 06 Park				
16:15		16:20	C 07 Terranova	16:10	C 12 Heitzenroeder		
16:30	C 01 Cole	16:40	C 08 Carralero	16:30	C 13 Goto		
16:50	C 02 Schmitt	17:00	C 09 Nishino	16:50	C 14 Schauer		
17:10	C 03 Lopez-Bruna	17:20	C 10 Otte	17:10	C 15 Garbedian		
17:30	C 04 Sugama	17:40	C 11 Sunn Pedersen	17:30	C 16 Bromberg		
17:50		18:00		17:50			
	Reception		NCSX & NSTX tours		Banquet		

Device Overview Posters (up all week)

P-D01	Mizuuhci	Overview of Heliotron J
P-D02	Otte	Overview of WEGA
P-D03	Bosch	Overview of W7-X
P-D04	Hidalgo	Overview of TJ-II
P-D05	Knowlton	Overview of CTH
P-D06	Talmadge	Overview of HSX
P-D07	Pedersen	Overview of CNT
P-D08	Yamada	Overview of LHD
P-D09	Blackwell	Overview of H-1
P-D10	Pankratov	Overview of Urugans
P-D11	Stroth	Overview of TJ-K
P-D12	Kitajima	Overview of TU-Heliac

Recent Results and Near-Future Plan of Heliotron J Project

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One of the major objectives of the Heliotron J project is to explore a concept optimization for the helical-axis heliotron, where bumpiness (ε_b) is introduced as a new parameter for the optimization. In recent Heliotron J experiments, we have focused on the effects of bumpiness on the plasma performance. Recent bumpiness control experiments indicate the following.

(1) To study the ε_b-effect on the bulk plasma confinement, following to the pervious study for ECH-only plasma, the enhancement factor of $\tau_E^{\mathrm{exp}}/\tau_E^{\mathrm{ISS95}}$ has been compared among the three configurations with different bumpiness (ε_b = 0.15 (high), 0.06 (medium) and 0.01 (low) at ρ = 0.67) for NBI-only plasma with the same line averaged density plasma. The higher enhancement factor has been obtained in the high- and medium-ε_b cases compared to that in the low- ε_b case. The improvement in T_i and T_e contributes to the higher enhancement factor in these configurations.

(2) To study the effect of the magnetic configuration on the generation and confinement of fast protons generated by ICRF minority heating, fast ion velocity distribution has been investigated. The high energy tail component extended to ~ 30 keV is observed near the pitch angle of 120° (the observation range: 111° - 128°) only in the high-ε_b case.

(3) A wide configuration scan shows that the EC driven current strongly depends on the magnetic ripple structure where the EC power is deposited. As the EC power is deposited on the deeper ripple bottom, the EC driven current flowing in the Fisch-Boozer direction decreases, and the reversal of directly measured EC driven current is observed. The normalized ECCD efficiency is found to be independent on the absorbed EC power for both ripple top and bottom heating cases. In order to increase the controllability of ECCD, the launching position and system has been changed.

Moreover, we have a lot of interesting observations (GAE activity and its effects on fast ion transport, structure of edge turbulence and its relation to the transition, non-local response caused by SMBI pulses, etc.) by using new diagnostics and/or a new fueling system (Supersonic Molecular Beam Injection). Some of them will be reported in this workshop.

In order to deepen the understanding not only of configuration effects on the plasma performance but also of enhanced confinement physics, our experimental plan (including under discussions) is as follows:

1. Expansion of investigation range in ($\varepsilon_t/\varepsilon_h$, $\varepsilon_b/\varepsilon_h$)-space with different iota values,
2. Build-up of profile database by improving the diagnostic system,
3. Expansion of achievable plasma parameter range by fueling and PWI control,
4. Increase of the plasma current controllability,
5. Comprehensive study of plasma turbulence.

Overview of Experimental Results from the WEGA Stellarator

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WEGA is a classical five period and $l = 2$ stellarator with a major radius of 72 cm and a maximum plasma radius of 11cm. Furthermore, WEGA is equipped with a set of vertical field coils which can be used for adjusting the radial position of the plasma and for varying the magnetic shear. With an additional error field compensation coil the machine has a very flexible magnetic configuration. Quasi stationary operation at 0.5 T for about 20 s is possible. The vacuum magnetic field structure has been intensively studied using the fluorescent technique and by a visualization technique based on the inelastic interaction of an electron beam in a background gas. Plasmas in hydrogen, argon and helium can be produced by two microwave heating systems operating at a frequency of 2.45 GHz (26 kW cw) and 28 GHz (10 kW cw). In addition WEGA can make use of an iron OH-transformer with a capacity of 440 mVs.

At 0.5 T operation two 28 GHz ECR heating scenarios have been applied: Second harmonic X2 mode ECR heating with central power deposition resulting into peaked plasma profiles with densities up to the cut-off of 5×10^{18} m^{-3} and electron temperatures in the order of a few ten eV. Furthermore, fully electron Bernstein wave (EBW) sustained plasma operation based on a two step mode conversion from electromagnetic O- to X-mode wave and succeeding X- to Bernstein wave was achieved with densities above the threshold of 1.0×10^{19} m^{-3}. While the bulk temperature of the electrons is of the order of a few ten eV, the EBE-diagnostics measured an extremely high radiation temperature in the range of a few ten keV in the plasma centre which could be confirmed by a soft X-ray diagnostic based on a pulse height analyser. This supra-thermal electron population is assumed to be generated by efficient EC-absorption of EBWs.

WEGA is equipped with several diagnostics. The line averaged density is measured by a single channel 80 GHz interferometer. Radiation profiles can be measured by a 12-channel bolometer camera and a 16-channel diode array. The non-absorbed 28 GHz ECRH stray radiation can be measured by a sniffer probe. A 12-channel radiometer (23 - 40 GHz) has been installed which is used for ECE and electron Bernstein wave emission (EBE) measurement. The continuous spectrum can be detected with a spectrum analyser during steady state plasma operation. Further contactless diagnostics are a heavy ion beam probe (HIBP), various spectrometers, video cameras and different Rogowski coils. For the determination of basic plasma parameter profiles a single Langmuir probe is installed on a fast reciprocating manipulator. For 0.5 T operation these results can be compared with HIBP measurements. Studies on turbulence and transport are possible with more sophisticated probe arrays. Special interest is paid to the influence of the magnetic field configuration on the dynamics of turbulent structures, including details on their parallel dynamics inside the scrape off layer (SOL). A 2D probe array is used to study turbulence in the region of magnetic islands which has been carefully characterized before by means of flux surface measurements.

Furthermore, on WEGA the prototype of the W7-X control system is realized and under continuous development. The new control system already demonstrated its potential by controlling a long-term discharge of 1 hour at low magnetic field.

CONSTRUCTION OF WENDELSTEIN 7-X: ENGINEERING A STEADY-STATE STELLARATOR

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The next step in the Wendelstein stellarator line is the large superconducting device Wendelstein 7-X. Presently under construction in Greifswald, Wendelstein 7-X is a “fully” optimised stellarator [1], based on the concept of quasi-isodynamicity. It has been designed for consistent operation of long-pulse (3600s) reactor relevant plasmas. The stellarator magnetic field and all components have been designed to ensure stable, high-beta steady state operation with good confinement of both the thermal plasma and fast ions, thereby demonstrating the reactor potential of optimised stellarators.

One key element of the Wendelstein 7-X mission is to demonstrate steady-state operation at reactor relevant plasma conditions, as required for an economic fusion reactor. Steady-state operation is an intrinsic feature of stellarators, unlike to tokamaks, where steady state operation is still a major challenge that requires extensive research and development efforts. Generally speaking, steady-state operation of fusion relevant plasmas is a complex task that is composed of both engineering and physics issues.

To be economically reasonable, the magnetic field in a fusion device has to be created with superconducting coils. The superconducting magnet system of Wendelstein 7-X consists of 70 superconducting coils and is one of the largest worldwide. Major components are 50 non-planar and 20 planar coils, a superconducting bus-bar system, 14 current-feedthroughs, a 360 channel quench detection system, and seven independent power supplies

Steady-state heating will be provided by a 10 MW ECRH-system at 140 GHz capable of 30 minute pulses. The 1MW gyrotrons have been newly developed to meet those requirements. The microwaves are guided to the torus with an optical transmission line.

For steady-state power exhaust, in total ten island-divertor modules will be installed in vessel. The island divertor concept has already proven successful in the predecessor device, Wendelstein 7-AS. The target elements are made of carbon fiber compound (CFC) and they are fully water-cooled to meet the design value of 10 MW/m^2 under steady state conditions.

This poster reviews the status of the construction of Wendelstein 7-X with a special emphasis on those components which constitute a genuine step in steady state technology towards a future fusion power reactor.

[1] J. Nührenberg, *et al.*, Trans. Fusion Techn. **27** (1995), 71

Overview of TJ-II experiments

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This paper presents the results on confinement studies in the TJ-II stellarator since the last ISHW Workshop. Considerable improvement of plasma particle control has been observed in the TJ-II stellarator after Li-coating, in comparison with the operation under Boron coated walls. The beneficial Li properties for plasma-wall interaction have a strong effect on this device that presents a helical limiter very close to the magnetic axis, which receives the strongest particle and heat fluxes. The outstanding results are the density control due to very low recycling conditions in formerly collapsing NBI discharges and the access to improved confinement regimes. A key ingredient for understanding the operational improvement is the change of profile radiation under Li coated wall. The edge radiation is observed to fall, which avoids the power unbalance that produces the low radiation collapse.

Confinement studies in ECH plasmas show that the lowest values for the effective electron heat diffusivity are found in regions where the lowest order magnetic resonances are located, while Alfven eigenmodes destabilized in NBI plasmas, also related to low order resonances, can influence fast ion confinement. A transition from kinetic effect-dominated to a more collisional regime is found in ECRH plasmas. The electric field, positive all over the plasma in the low ECRH plasma regime, starts developing negative values at the maximum density gradient region when the collisionality reaches a threshold value. For a given heating power and magnetic configuration, this translates into a line-density threshold to restore particle confinement. Further increments in the density extend the region with negative electric fields towards the center of the plasma.

During the high density NBI operation, a transition to an improved confinement regime is observed, characterised by the increase of diamagnetic energy, the decrease of Hα emission, the drastic reduction of turbulence, and the development of steep density gradients. High temporal and spatial resolution measurements indicate that turbulence reduction precedes the increase in the mean sheared flow, but is simultaneous with the increase in the low frequency oscillating sheared flow. So far, the H-mode has been obtained in a transient way and the estimated NBI absorbed power is comparable to the power threshold calculated using the empirical scaling obtained for tokamaks. This type of spontaneous transitions is added to the ones that happen at lower densities, which correspond to the shear flow development and can be also provoked by biasing. Regression analysis of the energy confinement time (up to 14 ms) indicates stronger degradation with power (power exponent – 0.8) and weaker density dependence (power exponent 0.4) than ISS04.

During both low and high plasma bifurcations, the correlation length of the plasma potential becomes of the order of the machine size during the edge bifurcation itself, quite unlike the density fluctuations. These results show that the increase in the degree of long-range correlation is strongly coupled to the presence of radial electric fields.

Future TJ-II experiments will be focussed on studying the efficiency of Electron Bernstein Waves (EBW) heating system using the OXB mode conversion scenario, physics of bifurcations and stability in high beta regimes and exploring plasma-wall interaction scenarios with Li coating and divertor concepts based on flux expansion.

Overview of the Compact Toroidal Hybrid experiment

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The Compact Toroidal Hybrid (CTH) experiment is a university-scale low-aspect ratio helical device with the distinguishing capability of operating with significant toroidal current such that the edge rotational transform due to the plasma current can be varied from zero to a value well in excess of the vacuum rotational transform. The experiment focuses on a greater understanding of current-driven disruptions in stellarators. While disruptions are typically not observed in helical systems, it is nonetheless of interest to investigate the MHD stability of stellarators with finite plasma current because disruption avoidance is relevant to helical configurations with tokamak-like levels of bootstrap current, e.g. quasi-axisymmetric devices, and stellarator-tokamak hybrids [1,2]. In close collaboration with the V3FIT equilibrium reconstruction code effort [3], experiments on current-carrying CTH plasmas also test new methods of three-dimensional plasma reconstruction of stellarator equilibria that are substantially different from the vacuum magnetic configuration. Furthermore, field-mapping studies have been used to experimentally determine subtle adjustments to the placement of the actual coils for an improved model of the as-built coil set. Further field error studies are planned for the future.

CTH (R_0 = 0.75 m, a ~ 0.2 m, $B_0 \leq 0.7$ T, $\bar{n}_e = 0.2 - 1.5 \times 10^{19}$ m^{-3}) is a flexible heliotron that, to date, operates with an ohmic current of $I_p \leq 45$ kA. At the standard field of $B_0 = 0.5$ T, plasmas are generated by electron-cyclotron resonant heating at 14 GHz, with an edge vacuum rotational transform variable from $\iota_{VAC}(a) = 0.04$ to 0.4. At the lowest vacuum transform, disruptions leading to a complete loss of plasma can be induced for total rotational transforms $\iota_{TOT}(a) > 0.3$ at plasma densities $\bar{n}_e \geq 0.8 \times 10^{19}$ m^{-3}. They are preceded by MHD activity observed on magnetic diagnostics and multi-channel soft X-ray cameras. Complete disruptions have not yet been observed at more typical operating scenarios in CTH with $\iota_{VAC}(a) \geq 0.1$. Initial reconstructions of current-carrying 3-D plasmas using magnetic diagnostics and V3FIT have been successfully performed, and will be used to characterize the plasma current profile for equilibrium and stability studies in ohmic CTH plasmas.

[1]. M. C. Zarnstorff et al., Plasma Phys. Controlled Fusion **43**, A237, 2001
[2] L. P. Ku and A. H. Boozer, Phys. Plasmas **16**, 082506 (2009)
[3] J. D. Hanson et al, Nucl. Fusion **49**, 075031 (2009)

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Overview of HSX Results

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HSX is a quasihelically symmetric stellarator with minimal toroidal curvature and high effective transform ($ɫ_{eff}$ = N – mɫ ~ 3). We present recent results at 1 Tesla operation which highlight the unique features of this configuration. An array of magnetic pick-up coils placed at two toroidal locations demonstrates that the Pfirsch-Schlüter (PS) current is helical. Meanwhile the bootstrap current is opposite in direction to that in a tokamak and both PS and bootstrap currents are reduced in magnitude by the high effective transform. Recently, first measurements of plasma flows using Charge Exchange Recombination Spectroscopy (CXRS) were made. Large parallel flows, on the order of 20 km/s were observed, in qualitative agreement with results first predicted by the PENTA code [1]. The neglect of parallel flows is a common and reasonable assumption for conventional stellarators when calculating the ambipolar electric field, but is demonstrated experimentally not to be valid for a quasihelically symmetric stellarator with low flow damping. Electron temperatures in the core during ECRH are up to 2.5 keV with 100 kW input power and drop to 1.5 keV when the symmetry is intentionally degraded and the neoclassical transport is increased. The steep temperature gradient in the core is indicative of a core electron root confinement (CERC) mode. PENTA calculations support the conclusion that even for a quasisymmetric stellarator with small symmetry-breaking, it is possible to achieve a neoclassical internal transport barrier based on the proximity of an electron root near an ion root. A Weiland ITG/TEM tokamak model for anomalous transport, which correct for the local geometry in HSX, supports the conclusion that E×B suppression of turbulence is responsible for the improved confinement in the plasma core [2]. At a lower field of 0.5 Tesla, an instability due to fast electrons produced by 2nd harmonic ECRH is observed. The mode is not present when the symmetry is broken. Experimental measurements of the mode frequency as a function of the rotational transform indicate that the mode is acoustic [3]. We will also briefly summarize topics of current research and future plans. Damping of zonal flows is expected to be reduced with quasisymmetry and is being explored with probes at the plasma edge. In addition, we have begun studies of how quasisymmetry affects impurity transport and the confinement of hot ions.

[1] D.A. Spong, Phys. Plasmas 12, 056114 (2005).
[2] W. Guttenfelder et al., Phys. Rev. Lett. 101, 215002 (2008).
[3] C.B. Deng et al., Phys. Rev. Lett. 103, 025003 (2009).

The Columbia Non-neutral Torus: Recent progress and future plans

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The Columbia Non-neutral Torus (CNT) is a small, compact stellarator dedicated to the study of non-neutral plasmas and electron-positron plasmas on magnetic surfaces. CNT has been in operation for nearly five years. Recent results include an experimental scan of the degree of neutrality from essentially pure electron (n_i/n_e<1%) to conditions approaching quasineutrality (n_i\approx n_e), improved understanding of both neutral driven transport and rod driven transport, an order of magnitude improvement in confinement time, and first results from operation without internal objects. One near term goal of CNT is to develop the ability to create and diagnose plasmas without internal objects. This is a necessary step before operation with positrons can commence. We will discuss our future plans for creation of electron-positron plasmas, including techniques to inject the positrons, and the need to accumulate cold positrons from a strong moderated source prior to injection.

Current Status and Nearest Future Plan of LHD

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The current status and the nearest future plan of the Large Helical Device (LHD) are summarized in this device report. The LHD is the heliotron-type device which employs the large scale superconducting magnets. The LHD has been operated quite stably without any serious troubles for 11 years and has provided more than 90,000 plasma discharges for a wide range of collaboration. The major goal of the LHD experiment is to demonstrate helical plasmas in a reactor relevant regime. Thorough exploration should lead to the establishment of not only a prospect for a helical fusion reactor but also to a comprehensive understanding of toroidal plasmas. Productive collaborating research on LHD has led to significant enhancement in plasma parameters of temperature, density, β and pulse length. In addition to this steady progress, new findings such as an Internal Diffusion Barrier (IDB) and Impurity Hole have brought breakthroughs to improve the prospects of a steady-state helical fusion reactor. Deep understanding and insight into the physical mechanisms attributed to a 3-D magnetic configuration have been built by our thorough study in LHD, which is also now recognized as a critical element in tokamaks. We completed the 12th experimental campaign in FY2008. The high ion temperature regime has been further extended with the achievement of the central ion temperature of 5.6 keV at the electron density of $1.6\times10^{19}m^{-3}$. Perpendicular NBI with a relatively low accelerating voltage of 40 keV is effective in ion heating and obtained high ion temperature is also associated with confinement improvement. The high density regime by an IDB and the high beta regime have been also extended steadily to $1.2\times10^{21}m^{-3}$ and 5.1 %. Systematic data in these frontiers have been accumulated, and new physical understanding and a feasibility of innovative scenarios are derived from this database. The extension of an operational regime has been conducted by 4 mission oriented theme groups in the LHD Experiment Group and 5 physics oriented theme groups have promoted keen approaches for specific as well as comprehensive understanding. An engineering issue, in particular superconducting technology, is also managed by a device engineering theme group. The 2nd perpendicular beam is under construction and will be available in 2010 and the total NBI power including 3 tangential NBI's will be 32 MW. The ion temperature could be scaled up to 8 keV by this new heating source. The vacuum vessel will be modified partially to install the proto-type closed divertor in 2010. The 13th experimental campaign is planned from Oct.2 to Dec.24 in 2009. The LHD is an experimental platform for cooperation by domestic and international fusion communities. Participation in collaboration is highly encouraged. The LHD Experiment Technical Guide is available at
http://www.lhd.nifs.ac.jp/lhd/databook_2009/LHD_technical_guide2009/.

The Australian National Plasma Fusion Facility: Results and Upgrade Plans – Device Overview Poster

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The 2009 Australian Budget included an allocation of approximately $US5M for the upgrade of the H-1 National Plasma Fusion Research Facility. Enhancements to the Facility will enable future growth of Australian capability in fusion science and engineering, and as a focus for collaboration within the Australian community, will support the development of world-class diagnostic systems for application to international facilities in preparation for ITER. The upgrade will include new heating systems and will deliver access to magnetic configurations more relevant to development of edge and divertor plasma diagnostics for next generation devices. The aims and implementation of the upgrade will be discussed in relation to the "Strategy for Fusion Science and Engineering in Australia"[1], developed by the Australian ITER Forum in consultation with the local plasma fusion community.

New results from some of the optical imaging and magnetic diagnostics[2,3] underpinning the upgrade plans will be presented, including a new method of coherence imaging of ion temperatures and flows. Imaging MHD mode structure using a synchronously-gated intensified optical emission camera has produced very encouraging initial data and together with data from two poloidal arrays of Mirnov coils and a precision step-scanned interferometer promises to provide detailed information about radial and toroidal mode structure of global MHD eigenmodes. For example, density and magnetic probe data indicate gradual transition of the mode to sound-like behaviour near low-order rational surfaces. Comparisons with CAS3D modelling and preliminary results from the application of Bayesian techniques to MHD mode structure analysis will be reported. Finally, studies of magnetic islands under plasma conditions, calibrated against precision island mapping in vacuum, have shown both beneficial and detrimental effects on confinement.

Future H-1 plans include construction of a new toroidal Mirnov array and application of coherence imaging and synchronous imaging techniques to MHD mode identification on H-1. Some of these imaging diagnostic tools are presently being deployed for the study of edge plasma and divertor flows, and for neutral beam imaging on DIII-D, Textor and KSTAR.

[1] http://www.ainse.edu.au/fusion/iter/fusion_energy_strategy_for_australia.html
[2] Blackwell, B.D. et al., Proc. 22nd IAEA Fusion Energy Conference (2008) EX/P9-11
[3] Kumar S.T, Blackwell B.D., Nuclear Fusion, 49 (2009) 035001

URAGAN-3M AND URAGAN-2M TORSATRONS

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Uragan-3M and Uragan-2M teams
Presented by I.M. Pankratov

Uragan-3M. Uragan-3M is an $l = 3, m = 9$ small size torsatron with major radius $R_0 = 1$ m, average plasma radius $\bar{a} \approx 0.12$ m and toroidal magnetic field $B_\phi = 0.72$ T. The whole magnetic system is enclosed into a large 5 m diameter vacuum chamber; an open natural helical divertor is realized. The rotational transform at the plasma boundary is $\iota(\bar{a})/2\pi \approx 0.3$ and magnetic well is 14%. The working gas (hydrogen) is admitted into the chamber continuously to provide the pressure in the range of 10^{-6}- 10^{-4} Torr. The plasma is produced and heated by RF fields (frame type antenna) at the frequency 8.8 MHz ($\omega \Box \omega_{ci}$) in the multi-mode Alfvén resonance regime.

The main recent achievement. The role of fast ion losses in the edge radial electric field bifurcation and H-mode transition (ETB formation) in RF discharge plasma has been investigated [1]. The reduced edge fluctuations level and turbulent transport are observed. An evident link between spectral/statistical characteristics of plasma fluctuations in SOL and DPFs, on the one hand, and fast ion loss, on the other hand, has been revealed.

The future program. The investigation of ETB formation under different RF heating scenarios. Studies of contribution of fast electron loss to the diverted plasma flow up-down asymmetry.

Uragan-2M. Uragan-2M is a medium size $l = 2, m = 4$ torsatron type system with a small pith angle of helical windings and additional toroidal field coils ($R = 1.7$ m, $\bar{a} \approx 0.2$ m). The ultimate value of toroidal; magnetic field is 2.4 T. The magnetic configurations with rotational transform $\iota(\bar{a})/2\pi > 1/3$ in the region near the magnetic axis and with $\iota(\bar{a})/2\pi < 1/2$ for outer surfaces are of practical interest, the magnetic well is 4.3%. In the present toroidal magnetic field did not exceed $B_\phi = 0.6$ T. At present Uragan-2M is equipped with two compact RF frame antennas.

The main recent achievement. Closed magnetic surfaces were measured for different values of toroidal and vertical fields. These measurements were in good agreement with calculated ones. Studies of RF discharges for wall conditioning have been carried out [2]. Continuous RF discharges are sustained by the 1 kW RF oscillator in the frequency range 4.5-8.8 MHz. The continuous discharge is also combined with a pulse discharge with power of 50-100 kW, 5.6 MHz, the pulse length is 10-20 ms and the repetition rate is 2-5 pulses per minute. In combined discharge the time of wall conditioning shortens. Both discharges seem to be suitable for wall conditioning and have a prospect for use in superconducting torsatrons.

The future program. For high ($\bar{n}_e \gtrsim 10^{13}\,cm^{-3}$) density plasma production and heating by RF methods a new four strap compact antenna will be used [3]. With this antenna the periphery plasma heating is suppressed and there is no sensitive dependence on the plasma parameters. The same antenna could be used for ICRH.

Study of particle and heat transport in RF discharge plasmas with reduced helical ripple will be carried out.

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2. V.E. Moiseenko, P.A. Burchenko, V.V. Chechkin et al. Wall Conditioning RF Discharges in Uragan-2M Torsatron, 36th EPS Conference on Plasma Physics (2009, Sofia, Bulgaria) P5.199.
3. V. E. Moiseenko, Ye. D. Volkov, V. I. Tereshin, Yu. S. Stadnik Alfvén Resonance Heating in Uragan-2M Torsatron 2009 Plasma Phys. Reports **35 (**in press)**.**

TJ-K: Recent Results and Future Projects

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The torsatron TJ-K is operated with a low-temperature plasma. This enables probe measurements inside the plasma with high temporal and spatial resolution. At the same time, dimensionless plasma parameters, which govern e.g. turbulent processes, are similar to those in the edge region of fusion plasmas. Therefore, the experiments are relevant for fusion research.

Detailed measurements of the turbulent characteristics and comparisons with turbulence simulations have clearly identified drift waves to be dominant in the TJ-K plasma: density and potential fluctuations propagate in phase into the electron diamagnetic direction [1], the structures have a finite parallel wavelength [2] and magnetic fluctuations are $2-3$ orders of magnitude below the electrostatic ones [3]. Recently, the investigations have been extended to the transition region from closed to open field lines. The generation of intermittent structures (blobs) at the separatrix has been demonstrated [4]. Multi-probe arrays have been used to study the energy transfer between turbulent fluctuations at different scales in wavenumber space. Different analyses techniques have been applied to simulated and measured data. In all cases evidence for the dual turbulent cascade has been gathered [5] and it was shown that the energy transfer is non-local in character [6]. Furthermore, core-biasing experiments have been carried out to study the influence of sheared flows on turbulent transport. A transition to improved confinement has been observed, together with a reduction of broadband turbulence and a concentration of the fluctuating power in coherent modes [7]. During improved confinement zonal flow like potential structures have been measured with the probe array and it was shown that these flows extract power from small scale fluctuations [6].

The second focus of experiments on TJ-K is wave physics. By full-wave simulations and experiments it was shown that heating with 2.45 GHz microwaves occurs off-axis at the upper hybrid resonance. This leads to hollow electron temperature but peaked density profiles [8]. The same mechanism is also responsible for heating with 8 GHz at 0.3 T. A new high density regime has been discovered when 2.45 GHz power has been added to an 8 GHz discharge. The 2.45 GHz microwave is absorbed very efficiently even though no resonance is present inside the plasma. The regime can be sustained even after switching off the 8 GHz source. The investigations of this regime are ongoing and will be part of the future experimental program.

In 2010 the experiment will be upgraded in heating power and magnetic field. A 7.5 kW system at 14 GHz will be available and the heating power at 8 GHz will be doubled to 2 kW. For resonant heating the experiment has to be upgraded to 0.5 T operation. This will allow us to access plasmas at lower collisionality with the objective to study neoclassical effects. As a preparation, measurements of the poloidal and toroidal flow velocities as well as the equilibrium currents have started. With respect to turbulence the upgrade will allow us to study turbulence at smaller scales which will be more relevant when comparing with fusion devices. In addition, the investigation of the influence of geometrical effects on turbulent transport will be a main focus of future experiments with the objective to compare turbulent growth rates with linear and non-linear simulations.

Ackn.: This work is financially supported by MPI für Plasmaphysik, EURATOM ASS.

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[2] N. Mahdizadeh et al., Plasma Phys. Contr. Fusion **49**(2007)1005
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[8] A. Köhn et al., Plasma Phys. Contr. Fusion submitted

Tohoku University Heliac (TU-Heliac)

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Tohoku University Heliac (TU-Heliac) is a 4-period heliac (major radius, 0.48 m; average plasma radius, 0.07 m). The heliac configurations were produced by three sets of magnetic field coils: 32 toroidal field coils, a center conductor coil, and one pair of vertical field coils. Three capacitor banks consisting of two-stage pulse forming networks separately supplied coil currents of 10 ms flat top. TU-Heliac group focuses the research projects on the following 4 subjects concerning the electrode biasing. The target plasma for the electrode biasing was He plasma produced by low frequency joule heating (f = 18.8 kHz, P_{out} ~ 35 kW). The joule heating power was supplied to one pair of poloidal coils wound outside the toroidal coils. The typical plasma parameters before biasing were as follows. The electron density on the magnetic axis was 6×10^{17} m^{-3} and the electron temperature on the axis was about 20 eV. The magnetic field on the axis was 0.3 T.

(1) Study of the relation between the magnetic ripple structure and the ion viscosity

We carried out the electrode biasing experiments and controlled the radial electric field and the poloidal flow. We examined the role of ion viscosity for transitions (flow bifurcation) and showed that the ion viscosity depends on the ripple structures in the magnetic configurations [1].

(2) Study of the density collapse in the improved confinement mode sustained by the electrode biasing

The density collapse was observed in the improved confinement mode sustained by the hot cathode biasing. The density profiles showed the steep gradient around the core plasma region before the collapse. The steep density profile collapsed accompanied with the bursting high frequency fluctuation ($100 < f < 400$ kHz), which had $m = 2$ poloidal mode number and the frequency agreed well with the $\boldsymbol{E} \times \boldsymbol{B}$ plasma rotating frequency [2].

(3) Research and development of a new type electrode for the biasing

For the development of a new field in biasing experiments, we have fabricated the new type electrode made of hydrogen storage metal for the particle injection (electron, ion and neutral particle). The high-density plasma ($>10^{19}$m^{-3}) was produced and the beta value increased up to about 0.5 % using the new type electrode made of gold (Au)-coated palladium (Pd) [3].

(4) Study of the rotating magnetic islands effects on the biased plasma

New method for the rotating magnetic islands by the external perturbation fields was proposed in TU-Heliac. The perturbation fields were produced by 4 pairs of cusp field coil. The phase difference in the floating potential signals measured by the two Langmuir probes confirmed that the magnetic islands rotated in the ion diamagnetic direction. These experimental results suggest the ability of the producing plasma poloidal rotation driven by rotating islands [4].

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[2] Y. Tanaka *et al.*, Plasma Fusion Res. **3** (2008) S1055.
[3] H. Utoh *et al.,* Fusion Sci. Tech. **50** (2006) 434.
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Monday

Oct. 12, 2009

08:45 – 09:00	Opening		
09:00 – 09:30	I01	H.S. Bosch	*Physics Programme for initial operation of Wendelstein 7-X*
09:30 – 10:00	I02	B. Blackwell	*The Australian National Plasma Fusion Facility: Results and Upgrade plans*
10:00 – 10:30	I03	I.M. Pankratov	*Studies of RF discharge plasma behaviour in the Uragan torsatrons*

Coffee

11:00 – 11:30	I04	P. Helander	*On plasma rotation and bootstrap currents in stellarators*
11:30 – 12:00	I05	J. Lore	*Neoclassical electron transport barrier in the HSX stellarator*
12:00 – 12:30	I06	S. Ohdachi	*Density collapse events in IDB/SDC plasmas on LHD*
12:30 – 13:00	I07	M. Yoshinuma	*Observation of impurity hole on LHD*

Lunch

14:00 – 16:00	P1-01	B.A. Stevenson	*Reconstruction of current-driven equilibria in the Compact Toroidal Hybrid* ***using*** …
14:00 – 16:00	P1-02	G.J. Hartwell	***Soft X-Ray Diagnostics on the Compact Toroidal Hybrid Experiment***
14:00 – 16:00	P1-03	S. Knowlton	***Disruptions in current-driven discharges in the Compact Toroidal Hybrid experiment***
14:00 – 16:00	P1-04	A. Briesemeister	***Initial Flow Velocity Measurements Using CHERS on HSX***
14:00 – 16:00	P1-05	R. Wilcox	***Zonal Flow Studies in the HSX Stellarator***
14:00 – 16:00	P1-06	Likin	*Confinement of fast ions in the HSX stellarator*
14:00 – 16:00	P1-07	K. Zhai	***Zeff Measurement from Bremsstrahlung Radiation on HSX***
14:00 – 16:00	P1-08	C. Clark	***A Computational Study of Impurity Transport with PENTA***
14:00 – 16:00	P1-09	B.Ph. van Milligan	***Tracer transport studies in simulation and experiment***
14:00 – 16:00	P1-10	J.A. Romero	***A half field configuration for high beta operation and plasma wall interaction studies…***
14:00 – 16:00	P1-11	F. Castejon	***ECRH-induced convective flux in TJ-II stellarator***
14:00 – 16:00	P1-12	J.M. Garcia-Regana	***Experimental characterization of the BXO Electron Bernstein Emission in the…***
14:00 – 16:00	P1-13	L. Krupnik	***Development of the bea, probe diagnostics for electric and magnetic field investigations …***
14:00 – 16:00	P1-14	L. Krupnik	***Installation and calibration of the heavy ion beam probe on WEGA stellarator***
14:00 – 16:00	P1-15	L. Krupnik	***Recent measurements in the TJ-II stellarator by HIBP diagnostic of the electric potential …***
14:00 – 16:00	P1-16	R. Ikeda	***Production and electron heating of over-dense plasmas by 2.45 GHz electron Bernstein …***
14:00 – 16:00	P1-17	T. Mizuuchi	***Effects of gas-fueling control on plasma performance in Heliotron J(***
14:00 – 16:00	P1-18	K. Mukai	***Electron density profile measurement in Heliotron J with a microwave AM reflectometer***
14:00 – 16:00	P1-19	K. Nagasaki	***Study of ECCD Physics and Iota Profile Control in Heliotron J***
14:00 – 16:00	P1-20	W. Kernbichler	***On ECCD efficiency in toroidal plasmas with finite collisionality***
14:00 – 16:00	P1-21	M. Bitter	***Conceptual Design of an X-ray Imaging Crystal Spectrometer for the Large Helical Device***
14:00 – 16:00	P1-22	X. Sarasola	***First observations of partially neutralized and quasineutral plasmas in the CN T***
14:00 – 16:00	P1-23	P.W. Brenner	***Confinement of Pure Electron Plasmas in the Columbia Non-neutral Torus***

16:00 – 16:30	I08	T. Morisaki	*Effect of Nonaxisymmetric Perturbation on Profile Formation*
16:30 – 16:50	C01	A.J. Cole	*Non-ambipolar particle fluxes and neoclassical viscosity in quasi-symmetry*
16:50 – 17:10	C02	J.C. Schmitt	*Modeling and Measurement of Toroidal Currents in the HSX Stellarator*
17:10 – 17:30	C03	D. Lopez-Bruna	*Local effects of magnetic resonances in ECRH plasmas of the TJ-II Heliac*
17:30 – 17:50	C04	H. Sugama	*Enhancement of Residual Zonal Flows in Helical Systemswith Equilibrium Radial …*

Physics Programme for initial operation of Wendelstein 7-X

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The Wendelstein 7-X stellarator, presently under construction in Greifswald, will be the first "fully-optimized" stellarator device which combines a quasi-symmetric magnetic field configuration with superconducting coils, a steady state exhaust concept, steady state heating at high power and a size sufficient to reach high nT τ-values. It is the mission of this project to demonstrate the reactor potential of the optimized stellarator line. A very important aspect in this is the demonstration of high power steady-state operation which is an intrinsic feature of stellarators.

After completion of the Wendelstein 7-X construction, scheduled for summer 2014, a staged approach for the scientific exploitation has been foreseen, starting with the commissioning phase which will take about a year. In the first operations phase, starting in summer 2015, basic stellarator properties and optimisation procedures are investigated with plasma discharges of 5 - 10 s duration. During this phase which should last no longer than 2 years, only a limited set of in-vessel components will be operational, i.e. a temporary divertor unit (TDU) with inertially cooled target plates will be installed and also the wall protection element will not be water-cooled.

Since the demonstration of steady-state operation is the substantial goal of W7-X, the initial commissioning and physics exploration phase with short discharges will be followed by a shut-down. During this shut-down the device will be completed to become steady-state capable. The TDU will be replaced with a water-cooled steady-state divertor, the cryo pumps will be installed and water-cooling will be completed.

After steady-state completion of the device, the physics programme will be continued and extended. Moreover, technological issues of steady-state fusion device operation will be addressed. In this phase also the physics and technology questions in connection with ITER and DEMO can be addressed.

For this staged approach to steady-state operation of Wendelstein 7-X, a framework for a research programme has been developed recently and will be presented in this paper.

The Australian National Plasma Fusion Facility: Results and Upgrade Plans

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The 2009 Australian Budget included an allocation of approximately $5M for the upgrade of the H-1 National Plasma Fusion Research Facility. Enhancements to the Facility will enable the growth of Australian capability in fusion science and engineering, and will support the development of world class diagnostic systems for application to international facilities in preparation for ITER. The upgrade will include new heating systems and will deliver access to new magnetic configurations relevant to development of edge and divertor plasma diagnostics for next generation devices. The aims and implementation of the upgrade will be discussed in relation to the "Strategy for Fusion Science and Engineering in Australia"[1], developed by Australian ITER Forum in consultation with the plasma fusion community. The Facility research plan will be presented, including target parameters and configurations, modelling results, and ways in which the international community could be involved. New results from some of the optical imaging and magnetic diagnostics[2,3] underpinning the upgrade plans will be presented, including a new method of coherence imaging of ion temperatures and flows. Synchronous imaging of MHD mode structure using fast optical emission cameras promises to supplement data from two poloidal arrays of Mirnov coils and a precision step-scanned interferometer to provide detailed information about radial and toroidal mode structure. Comparisons with theory will include a CAS3D study and preliminary results from the application of Bayesian techniques to MHD mode structure analysis in H-1.

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[2] Blackwell, B.D. et al., Proc. 22nd IAEA Fusion Energy Conference (2008) EX/P9-11
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Studies of RF discharge plasma behavior in the Uragan-3M and Uragan-2M torsatrons

I. M. Pankratov, A. A. Beletskii, V. L. Berezhnyj, P. Ya. Burchenko, V. V. Chechkin, L. I. Grigor'eva, D. Hartmann[1], R. Koch[2], V. G. Konovalov, A. Ye. Kulaga, G. G. Lesnyakov, A. V. Lozin, A. I. Lyssoivan[2], S. M. Maznichenko, Yu. K. Mironov, V. Ye. Moiseenko, V. K. Pashnev, A. A. Petrushenya, A. V. Prokopenko, V. S. Romanov, A. N. Shapoval, A. F. Shtan', D. A. Sitnikov, A. I. Skibenko, A. S. Slavnyj, E. L. Sorokovoy, Ye. L. Sorokovoy, O. M. Shvets, Yu. S. Stadnik, I. K. Tarasov, V. I. Tereshin, S. A. Tsybenko, N. V. Zamanov, Ye. D. Volkov

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Results are reviewed of experiments on using RF fields for plasma production and heating and for wall conditioning by RF discharge plasmas in the U-3M and U-2M torsatrons.

In the l=3/m=9 U-3M torsatron (R=1 m, $\bar{a}\approx$0.12 m, $\iota(\bar{a})\approx$0.3, B_ϕ = 0.7 T) in the Alfven resonance heating regime, the processes of ITB and ETB formation are investigated. It is shown that occurrence of the transport barriers is associated with the island structure of the magnetic configuration (ITB) and generation of fast ions in the RF discharge and their loss (ETB).

Studies of RF discharges for wall conditioning have been carried out in the U-2M torsatron with reduced helical ripples and additional toroidal field coils (l=2, m=4, R=1.7 m, $\bar{a}\approx$0.22-0.24 m, $\iota(\bar{a})\approx$0.75, present B_ϕ ~ 0.6 T).

Continuous RF discharges in U-2M are sustained by the 1 kW RF generator in the frequency range 4.5-8.8 MHz. The RF power is launched to the plasma by a frame antenna. The discharge parameters are measured in a wide range of confining magnetic field, pressures and RF power. The state of wall conditioning is evaluated by the evolution of the impurities in the discharge signified by the optical measurements, the residual gas composition and partial pressures measured with the mass-spectrometer. This evolution is analyzed during several days of the U-2M operation.

The continuous discharge is combined with a pulse discharge with the power of 50-100 kW, 5.6 MHz, pulse length 10-20 ms, 2-5 pulses per minute. This shortens the time of wall conditioning. As a result, the possibility has occurred to realize RF plasma heating in the Alfven range of frequencies.

On plasma rotation and bootstrap currents in stellarators

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This talk will describe two recent exact results in kinetic stellarator theory.

First, the question of stellarator plasma rotation is explored [1]. It is shown that gyrokinetic turbulence is fundamentally unable to affect the large-scale rotation of a stellarator plasma, unless the magnetic field is very close to being perfectly quasiaxisymmetric or quasihelically symmetric. This conclusion follows from the very orderings underlying the derivation of the gyrokinetic equation: the turbulent transport of momentum, by the Reynolds and Maxwell stresses, is smaller than the parallel neoclassical viscosity. This viscosity clamps the plasma rotation (both poloidal and toroidal) and radial electric field at the value corresponding to ambipolar neoclassical transport. Only in quasisymmetric systems is the neoclassical transport intrinsically ambipolar and momentum transport plays a role in determining the radial electric field. On small scales, however, the Reynolds and Maxwell stresses may be large enough to drive zonal flows with radial wavelengths comparable to the ion gyroradius.

The second topic treated in the talk is neoclassical theory in a perfectly quasi-isodynamic (omnigenous) stellarator [2]. In such a device, there is of course no $1/\nu$-transport, so that the neoclassical confinement is tokamak-like. However, unlike the situation in a tokamak, it can be shown that the bootstrap current vanishes exactly in the limit of long mean-free path, if the trapped particles precess poloidally (rather than toroidally or helically). Two of the optimisation criteria for Wendelstein 7-X-like stellarators – low neoclassical transport and small bootstrap current – therefore actually coincide.

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[2] P. Helander and J. Nührenberg, Plasma Phys. Control. Fusion **51**, 055004 (2009).

Neoclassical Electron Transport Barrier in the HSX Stellarator

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Strongly peaked electron temperature profiles are measured in the core of the Helically Symmetric Experiment (HSX) during electron cyclotron heating; with central temperatures of 2.5 keV for 100kW of injected power. These measurements, coupled with neoclassical predictions of large "electron root" radial electric fields with strong radial shear, are evidence of a neoclassically driven thermal transport barrier.

Neoclassical transport is analyzed using the PENTA code[1], in which parallel momentum is conserved, based on moments methods developed by Sugama and Nishimura[2]. Momentum conservation, including the effects of parallel flow, has long been known to be important in tokamak neoclassical calculations. Conventional stellarators, on the other hand, typically exhibit strong flow damping in all directions on a flux surface, and the parallel flows can be neglected. In this case, the radial electric field is calculated using a simple ambipolarity constraint: setting the ion flux equal to the electron flux. In optimized stellarators with very low effective ripple, such as HSX, parallel flow and momentum conservation are again expected to be important. Large parallel flow measurements (~20km/s) from the Charge Exchange Recombination Spectroscopy (ChERS) diagnostic are consistent with reduced damping in the direction of symmetry.

In addition to neoclassical transport, a model of Trapped Electron Mode turbulence is used to calculate the turbulent-driven electron thermal diffusivity. The very peaked T_e profile is reproduced by predictive transport simulations only when turbulent transport quenching via sheared ExB flow is included[3]. ChERS measurements of the radial electric field profile and comparison to the neoclassical calculations will also be presented.

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Density collapse events observed in IDB/SDC plasmas on LHD

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Plasma with an internal diffusion barrier (IDB) is one of a promising configuration for the Heliotron-type reactor. Extremely peaked density profile is formed by the central fueling with ice-pellet injection [1]. To study the stability and sustainability of the IDB plasma is thus of great importance; MHD instabilities due to the fairy high-pressure-gradient in the barrier region are especially in concern. Experimentally, we observe so-called core density collapse (CDC) events where the peaked density profile is broken within 1ms when the central beta value and/or the magnetic axis position exceed a certain threshold value. The stored energy is decreased by several tens of percent by an event. The cause of CDC has not clarified yet. Here, we present a study based on an idea that CDC is caused by the pressure-gradient driven MHD instabilities.

Typical time constant of a CDC increases with a decrease of the toroidal magnetic field. The electron density in IDB plasmas is much lower in a lower magnetic field, whereas the electron temperature does not change as well. Thereby, by changing the magnetic field we can study the effects of the collisionality. With lower collisionality (or low magnetic field), the scale of an event becomes small and pre-cursor oscillations localized in the outboard side of the peaked profile are sometimes seen. Large-scale continuous oscillations (~1kHz) with a dominant poloidal/toroidal mode number m/n = 1/1 are observed when the magnetic field is further reduced to 1T or 0.75T under the same condition. Those pre-cursors and continuous oscillation are considered to be evidence that CDC is caused by the MHD instabilities. Characteristics of the CDC events and comparison with numerical analysis of the instabilities will be presented in detail.

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Observations of Impurity Hole on LHD

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Impurity hole, which is an extremely hollow profile of carbon impurity, is observed associated with increase of ion temperature gradient in the Large Helical Device. Neoclassical prediction can not explain the outward convection which causes the hollow profile. The experimental observation of the outward convection driven by the ion temperature gradient is a important finding because it gives a prospect for simultaneous achievement of the good energy confinement and low impurity confinement in non-axisymmetric systems.

The plasma with a steep ion temperature gradient is produced with a injection of cylindrical carbon pellet into the plasma sustained with neutral beam injection heating. The impurity hole, which is characterized by an extremely hollow profile of impurity density, is observed in the decay phase after the carbon pellet injection, where the ion temperature gradient increases due to both the increase of deposition power and the improvement of ion transport. The radial profile of the carbon density is peaked just after the carbon pellet injection and becomes extremely hollow in the time scale of a few hundred milliseconds during the decay phase. On the contrary, the electron density profile stays in peaked and does not become hollow. This observation indicates that the bulk ion profile is always peaked during the decay phase because of the fueling of the neutral beams. The hollow profile of the carbon density is due to the faster decay of the carbon density at the centre than that at the edge and not due to the increase of the influx. The gradient of ion temperature increases and reaches 10keV/m, while the gradient of electron temperature is weak (~3keV/m) during the decay phase of the electron density.

The diffusion coefficient and convection velocity of both carbon and proton are evaluated from the particle flux plotted as the function of the density gradient. The experimental data show the outward convection which is opposite to that predicted by the neoclassical theory (inward convection) in the plasma core region where the radial electric field is negative during the impurity-hole formation. Since the ion temperature gradient rather than the electron temperature gradient starts to increase at the beginning of the impurity-hole formation, the ion temperature gradient is the most possible candidate for driving the outward convection. The convection velocity of carbon increases as the ion temperature gradient is increased. When the impurity hole appears in the plasma, a strong drop in the concentration of the argon impurity is also observed, although the concentration of helium in the plasma shows only a gradual decrease. These results suggest the Z-dependence of the outward convection velocity, where the higher-Z impurity has a stronger outward convection. The strength of the impurity hole depends on the configuration of the magnetic field, especially the position of the magnetic axis. The impurity hole becomes stronger as the magnetic axis is shifted outward.

Reconstruction of current-driven equilibria in the Compact Toroidal Hybrid using magnetic diagnostics

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There is a growing need for rapid reconstruction capability of fully three-dimensional equilibria in toroidal confinement experiments. Test and validation of the new V3FIT 3D magnetic equilibrium reconstruction code [1] are underway on the Compact Toroidal Hybrid (CTH). The CTH is a heliotron-type device in which the magnetic configuration can be strongly modified by plasma current. The operating parameters of CTH are $R_0 = 0.75$ m, a ~ 0.2 m, $B_0 \leq 0.7$ T, $n_e = 0.2 – 1.5 \times 10^{19}$ m^{-3}, $T_e \sim 200$ eV, $I_p \leq 40$ kA, $\iota_{vac}(a) = 0.05 – 0.4$. The present suite of magnetic diagnostics includes internal and external 8-part and full Rogowski coils, four flux loops, and a diamagnetic loop. The measured signals from these diagnostics include contributions from the plasma current, externally applied currents, vacuum vessel current, and various sources of pickup and drift. The induced toroidal vacuum vessel current (<15kA) significantly contributes to the magnetic diagnostic signals. In order to include this current contribution in the reconstruction process, the VALEN code [2] was used to model the time varying vacuum vessel current distribution. For reconstructions, the plasma contribution is extracted from the total signal to provide the experimental input to V3FIT which utilizes least-squares fitting and the VMEC equilibrium code [3] to reconstruct 3D plasma equilibria. Efforts are underway to compare the flux surface geometry reconstructed using magnetic diagnostics with those from soft X-ray tomography (see poster by Hartwell et al., this conference).

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Soft X-Ray Diagnostics on the Compact Toroidal Hybrid Experiment

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Soft X-Ray (SXR) Diagnostics are used on the Compact Toroidal Hybrid (CTH) torsatron experiment ($R = 0.75$ m, $a \sim 0.2$ m, $B \leq 0.7$ T, $n_e \leq 10^{19}$ m^{-3}, $T_e \leq 250$ eV) for tomographic reconstruction of the emissivity profile, electron temperature measurement, and as input to a 3D reconstruction code. SXR tomography is performed with three cameras with up to 60 chords viewing a poloidal cross-section. Each camera consists of a 20-channel AXUV-20EL photo-diode array filtered with 500nm Al foil. Electron temperatures are being measured with an Amptek X123-SDD spectrometer. The spectrometer views the Bremsstrahlung emission along a single chord through the plasma in the energy range from 1-10 keV. Signals from the 60 channel tomographic camera system and signals from similar cameras located at other toroidal locations are being incorporated into the V3FIT reconstruction code[1].

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Disruptions in current-driven discharges in the Compact Toroidal Hybrid experiment

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While disruptions are typically not observed in helical devices, it is nonetheless of interest to investigate the MHD stability of stellarators with finite plasma current. Understanding of disruption avoidance is relevant to helical configurations with tokamak-like levels of bootstrap current, e.g. quasi-axisymmetric devices, and stellarator-tokamak hybrids [1]. Disruptions in helical geometry are studied in the Compact Toridal Hybrid (CTH) experiment (R_0 = 0.75 m, a ~ 0.2 m, $B_0 \leq 0.7$ T, $\bar{n}_e$ = 0.2 – 1.5 x 10^{19} m^{-3}), a flexible heliotron with the capability of operating with significant ohmic current. At the standard field of B_0 = 0.5 T, the edge vacuum rotational transform is variable from $\iota_{VAC}(a)$ = 0.05 to 0.5 and plasma currents up to 40 kA are driven in plasmas generated by electron-cyclotron resonant heating at 14 GHz. At the lowest vacuum transform $\iota_{VAC}(a)$ = 0.05, current-driven disruptions leading to a complete loss of plasma can be induced for total rotational transforms $\iota_{TOT}(a) > 0.3$ at plasma densities $\bar{n}_e \geq 0.8$ x 10^{19} m^{-3}. They are preceded by what is presently interpreted to be m = 2 activity observed on magnetic diagnostics and multi-channel soft X-ray cameras. Complete disruptions have not been observed at more typical operating scenarios in CTH with $\iota_{VAC}(a) \geq$ 0.2, although partial current decreases often take place. Efforts are underway to study the transitional behavior of the disruptive activity as the vacuum transform is continuously raised from its lowest value, and to model the pre-disruptive plasma equilibrium with the V3FIT 3D equilibrium reconstruction code [2].

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Initial Flow Velocity Measurements Using CHERS on HSX

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Flow velocity measurements made using Charge Exchange Recombination Spectroscopy (CHERS) on the Helically Symmetric Experiment (HSX), a quasi-helically symmetric stellarator are presented. Measurements have been performed in 1Tesla, O-mode ECRH heated plasmas, with on-axis electron temperatures of up to 2keV, and very peaked temperature profiles which indicate enhanced core electron root confinement (CERC). The CHERS measurements are made using the 529nm C^{+5} emission line. A 30keV, predominantly mono-energetic, 4 Amp-equivalent, 3ms neutral hydrogen beam is employed. Two .75m Czerny-Turner spectrometers with electron-multiplying ccds are being used. The system has a dispersion of about 0.06Å/pixel. The system is calibrated during every shot using a Ne pencil style calibration lamp. The use of two viewing angles at 10 radial locations allows the flow direction in addition to the magnitude to be measured. One set of views is oriented almost perpendicular to the direction of symmetry and tangent to the flux surface (an approximately poloidal view), while the other set of views is oriented almost parallel to the symmetry direction (an approximately toroidal view). The Doppler shift is measured during a series of shots with the magnetic field in the counterclockwise direction as well as for a series of shots with the field in the clockwise direction. The change in wavelength from shots with different B field directions is used to find the flow velocity. During quasi-helically symmetric operations measurements confirm that the plasma flow velocity is predominantly parallel to the direction of symmetry with a peak magnitude on the order of 20km/s. Velocity measurements along with force balance are used to calculate the radial electric field. A comparison of the measurements to a momentum-conserving neoclassical calculation of the flow velocity and radial electric field will be presented.

Zonal Flow Studies in the HSX Stellarator

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The distinguishing feature of HSX is that there is symmetry in the magnitude of the magnetic field in the helical direction. One implication of quasi-symmetry that has been predicted and measured is the reduction of viscous flow damping in the direction of symmetry. The effect that this has on the evolution of zonal flows with and without an induced bias is to be investigated. Measurements of correlations between density and potential fluctuations as indicators of zonal flows are presented. Biasing experiments are also conducted, with the bias applied at the plasma edge relative to a carbon limiter placed near the last closed flux surface. The goal is to induce a radial electric field beyond the value of the poloidal resonance (where the ExB velocity cancels the poloidal motion of an ion) where the transport and nonlinear viscosity is predicted to be significantly reduced. Initial measurements of density and potential fluctuations with the bias applied are presented.

Confinement of fast ions in the HSX stellarator

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Our initial results on single particle orbits in the HSX show a strong dependence of fast ion confinement on a presence of small higher harmonics in the mod B spectrum. In those calculations a standard VMEC output spectrum was used. To be certain about the amplitude of the high harmonics we have compared the VMEC spectrum and the spectrum from a field line following code. The trajectories of fast ion orbits distributed over multiple spatial locations and pitch angles were calculated to present a map of the loss orbit locations in HSX. The magnetic geometry was then varied to minimize the loss orbit regions. Minority ion D(H) heating in HSX has been modelled with the GNET code which solves the drift kinetic equation in 5-D phase space [1]. Efficiency and particle flux driven by the ion cyclotron resonance heating (ICRH) are presented in this paper. A possible scenario to achieve a negative radial electric field (“ion root”) for HSX plasmas is also discussed.

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Zeff Measurement from Bremsstrahlung Radiation on HSX

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Zeff measurements using plasma bremsstrahlung radiation on the Helically Symmetric Experiment (HSX) are being developed. The system uses the HSX CHERS optics which consist of collection optics, a coupling optical fiber bundle, a 0.75m spectrometer with 2400g/mm grating, and an EMCCD. The instrument measures the line integrated brightness from ten chords through the HSX plasma in the wavelength region near 529nm. The free-bound recombination radiation is negligible at this visible wavelength region which is chosen to be free from the plasma line radiation and thus is dominated by bremsstrahlung radiation. Absolute intensity calibration is done with a calibrated integrating sphere. Through reconstruction of the line averaged signal, a Zeff profile can be inferred. Detailed experimental setup and the initial results will be presented.

A Computational Study of Impurity Transport with PENTA

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Neoclassical impurity transport will be studied using the PENTA code [1] which conserves parallel momentum and accounts for parallel flows. The transport coefficients that are generated by DKES [2] are calculated using a non-momentum conserving collision operator, but PENTA modifies them to conserve parallel momentum using a moments method [3]. Modifications allowing PENTA to couple a user defined set of ion species have been made to enable impurity transport analysis. This study will focus on the convective portion of the impurity transport, and specifically on the transport driven by the ion temperature gradient, which can produce a "temperature screening" effect in axisymmetric devices, but not in general 3D devices.

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Tracer transport studies in simulation and experiment

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The study of the transport of ideal or actual tracer particles allows obtaining information about the underlying transport mechanisms in plasmas that cannot be obtained by other methods. In particular, it is possible to test whether transport is local and Markovian or not. To a large degree, this is still an open issue.

In this work we will present analysis techniques for the detection of non-local and non-Markovian behavior and results from their application in various simulations. Firstly, we will discuss a comparison of such analyses in an L-mode and an H-mode run of the fluid turbulence code CUTIE [1]. Secondly, we will present results from the gyrokinetic code UCAN, focusing on the effect of sheared flow on transport [2].

Finally, we will discuss the perspectives for applying these techniques in experiments, and present an experiment based on the test particle approach that will be performed in the Large Helical Device (LHD) in Japan this autumn. In order to inject the physical tracers locally into the high-temperature plasma, a solid pellet with tracers inside a protective outer shell will be used (TESPEL) [3]. Tracer dispersion will be followed using a fast camera.

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A half field configuration for high beta operation and plasma wall interaction studies at TJ-II

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The standard configuration space of the TJ-II heliac is dominated by an strong plasma wall interaction. Plasma is only a few centimeters away from the hardcore protection tiles, and in some configurations is limited by them. To reduce the plasma wall interaction, plasma has to be place further away from the hardcore. The distance of the plasma to the hardcore depends of the ratio between poloidal and toroidal field components. Increasing the poloidal field is limited by engineering limits on the central and helical conductor coils current. So the only possibility to move the plasma away from the hardcore is to reduce the toroidal magnetic field. This has to be done in a way compatible with the existing plasma heating mechanisms available. TJ-II has two 300kW, 53.2 GHz gyrotrons with a second harmonic resonance at 0.96T. The new experimental campaign, TJ-II will have an additional 300kW, 28 GHz gyrotron aiming to Electron Bernstein Wave (EBW) heating system at first harmonic at 1T . The EBW system could also be used as a standard second harmonic heating at 0.5 T. In addition, TJ-II has two NBI injection systems with port-through powers between 200 and 400kW and injection energy 28 kV. So we have 300kW ECH and about 600kW NBI available for half field configuration heating. In the search for a half field configuration with large distance to the hardcore, we have found an interesting operational space of half-field configurations with large volume plasmas (1m3). These configurations can be used to study heliac plasmas with reduced plasma-wall interaction and explore high beta operation using the toroidal field reduction.
We have selected one particular half field configuration whose LCFS is placed more than 12cm away from the inner hardcore walls. This is expected to reduce the plasma wall interaction providing an extra leverage for density control in NBI discharges.
Neutral Beam transmission is limited by beam interception at the injection port and the first toroidal field coil. Beam steering optimization is of critical importance. Beam power absorption as function of injection angles has been studied using the Monte Carlo simulation code FAFNER. The beam trapping profiles are obtained, as well as the corresponding shine through losses, which are mainly dependent on the plasma density profile. The fast ion trajectories are followed in the 3D magnetic configuration of TJ-II, and the energy transfer to the plasma electrons and ions is computed for different sets of temperature and density profiles. In addition to the shine through losses, there are fast ion losses due to bad orbits and losses due to charge exchange of the fast ions with the neutral gas molecules present in the vacuum chamber. The best injection occurs when the NBI beam targets the magnetic axis. Due to the small size of the available injection window, this angle is not accessible without large power losses in the NBI duct walls, toroidal field coil TF1 protection and window surrounding structures. The absorbed power in the 0.5T configuration is 57% of the absorbed power in the 0.96T reference configuration. The remaining question is whether the expected reduction in plasma wall interaction can overcome the degraded beam absorption. The new configuration space could be used to study plasma wall interaction as function of distance to the hardcore, and high beta operation using half field configurations.

ECRH-induced convective flux in TJ-II stellarator

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Fast transport phenomena in plasmas are usually denoted as non-local transport to show that the perturbation is propagated much faster than the background diffusive transport. As the perturbations propagate very fast, it looks like the perturbations propagated instantaneously, especially when the time scale of the propagation is faster than the time resolution of the diagnostics that are used. If one studies larger time scales, the fluxes must be treated as dependent on the plasma characteristics of points far from the one where the transport is estimated [1], giving rise to global-like phenomenon. Beyond the formal description of the non-local transport, it would be necessary to give its dynamical explanation. Both heat and cold pulse propagation have exhibited this kind of behaviour, but a possible explanation based on kinetic effect is proposed here only for heat pulse propagation.

Heat wave experiments have been used in tokamaks and stellarators (see e. g. [2] and [3]) to study electron heat transport and to estimate the power deposition profile. Low modulation frequencies are used to study transport, while high frequencies are to determine the power deposition profile. The latter experiments are based on the fact that bulk electrons are not able to follow the fast power cycles of the modulation. Nevertheless, fast transport phenomena should be taken into account in order that the measurements are interpreted properly.

We present in this work the results of fast and slow modulation experiments that show several effects that cannot be attributed to a diffusive behaviour, namely convective transport and a systematic widening of power deposition profile in comparison with the predicted by WKB theory for a Maxwellian distribution function. The former results are related to the appearance of an extra outward electron flux induced by ECRH both in tokamaks and stellarators. Some experimental features of the existence of this flux are the hollow density profiles and the increase of H_α emission when the gyrotron is switched on, as well as the onset of a suprathermal component in SXR spectra and an increase of SXR flux emissions [4]. Such a convective transport has been characterised and its consequences on non-local transport description as well as on power modulation are discussed. The results shown here are compared with the previous calculations presented in [5].

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Experimental characterization of the BXO Electron Bernstein Emission in the TJ-II Stellarator

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The Electron Bernstein Waves (EBW) heating system of the TJ-II Stellarator (OXB mode conversion scenario at 1st harmonic) is provided with an internal steerable mirror that makes possible the experimental optimization of the launching direction [1]. Moreover, an Electron Bernstein Emission (EBE) diagnostic [2] was installed to measure the BXO mode converted radiation in overdense plasmas obtained with NBI discharges. The internal mirror of the launching system is also used to focus the radiation towards the detection antenna. Therefore, since the line of sight of the radiometer matches the power injection direction of the heating system, the optimum heating position may be determined from the analysis of the measured radiation.

A mirror position scan has been performed looking for the mirror orientation angles that maximize the intensity of the detected O mode radiation. Preliminary analysis, in rough agreement with the theoretical estimations (see figure 1), carried out using the TRUBA code [3,4], was presented in [5].

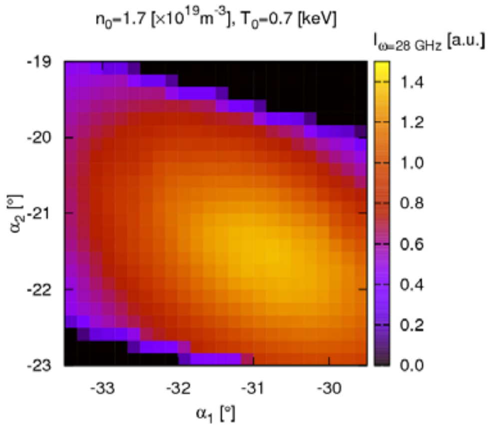

Figure 1: Theoretical Electron Bernstein Emission window in TJ-II Stellarator for central electron density and temperature values of $n_0=1.7\times10^{19}$ m^{-3} and $T_0=0.7$ keV. α_1 and α_2 are the mirror positioning angles.

Here, a detailed study of the available data, both in ECRH and NBI plasmas, is presented. The results provide valuable information to achieve an optimum O-X coupling in the EBW heating experiments. The EBW heating system will come into operation during the fall of 2009.

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DEVELOPMENT OF THE BEAM PROBE DIAGNOSTICS FOR ELECTRIC AND MAGNETIC FIELDS INVESTIGATIONS FOR URAGAN-2M

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To investigate of the plasma electric and magnetic fields in devices with magnetic confinement it was given preference to Beam Probe Diagnostics

Plasma probing by neutral and ionic atomoc beams give ability to obtain information about space distribution of an electric potential, plasma density, electron temperature, impurity ions space distribution and poloidal magnetic field (plasma current) in the hot plasma of existing modern thermonuclear devices.

Up to now it is known two branches of beam probe diagnostic:

- a heavy ion beam probing (Cs+, Tl+) (HIBP)
- the light neutral atomic particle probing (Li,He,H) (BES,CXS,MSE).

In this report will be presented novel elaborations in HIBP and BES diagnostics fulfilled in the Kharkov IPP for Ukranian stellarator Uragan 2M.

U2-M is the flexible torsatron with small helical ripples and considerably high parameters (R_0 = 170 cm, ape = 22 cm, B_0 = 0,8 – 2,4 T, l = 2, m = 4). It was put to operation at the end of 2006.

It was been optimized, manufactured and tested a new HIBP and BES diagnosic sets to make investigation of the electric and magnetic fields of this device.

The basic problem for realisation of the poloidal magnetic fields measurements by Beam Emission Spectrometry is a low level of an optical signal determined by value of probing atomic stream. The simplest way to solve this problem - is of a primary ion beam current increasing. For U-2M it was been developed an ijnjector with parameters: accelerating voltage up to 70 kV, light alkali ion current (Li+, Na+)up to 15-20 mA., emitter temperature up to 1500 C.

The calculations of the HIBP trajectories for U-2M were made using singly charged cesium and thallium(Cs+ and Tl+) primary ions in the energy range from 150keV to 950 keV

Two variants of HIBP diagnostic for different magnetic field values were calculated. At the first stage of the stellarator’s operation toroidal magnetic field will be 0.8 T, at the second stage – up to 2.4 T. Detector grid, that was calculated, covers quite large area of the plasma. It is possible to get the plasma potential profile by fast electrostatic deflection scanning system in the range of 0.1<□<1. For the first stage Tl+ beam of 150 keV will be more than enough. Thye second stage needs to increase of the Tl+ beam energy up to 950 keV.

On the basis of trajectories calculations the diagnostic complex has been developed. It traditional consists from injector of the probing beam and the secondary particle analyser. The high-voltage power supply has been developed for a feed of the first stage of the stellarator operation with a voltage up to 200 □V and a current 1 □A. Industrial power supply IVN-100 (100 □V, 1 □A), will be used for a feed of the analyzer.

INSTALLATION AND CALIBRATION HEAVY ION BEAM PROBE ON WEGA STELLARATOR

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The conceptual design for a Heavy Ion Beam diagnostic (HIBP) for the stellarator WEGA in Greifswald (Germany) is developed to provide the measurements of the radial profiles of the electric plasma potential, density and their fluctuations. Calculations of probing Na+ beam trajectories were done for the various WEGA diagnostics ports with B_0=0.5T. The information about the plasma parameters is obtained from sample volume which is moving along the detector line during scanning the deflecting voltages in a primary and secondary beam lines. The covered radial range is $0.4 < r/a < 1$, of effective radius. The plasma centre (r/a=0) is not accessible due to geometrical limitations.

Different size of the sample volume leads to a different sensitivity and resolution in a different parts of measured profile. In our case at the edge it is more sensitive, but has lower resolution than is the centre.

The very important and most difficult to define is the coordinate of the sample volume. Up to now only information we had about it was obtained from ray tracing calculations. However, the magnetic field and geometry of WEGA and HIBP in calculations is idealized and could differ from that in reality.

To implement the corrections into a calculations we need more information about the real beam position in WEGA. To provide this information the detector array was installed in the HIBP cross section for the primary beam position measurements. Deviation of calculations relative to measurements is negligible. The reasons for this deviation assumed to be an inaccuracy of the HIBP assembly. Especially the deflecting plates positions and primary beam accelerator. And magnetic field of the WEGA calculation.

The HIBP on WEGA is planned to be used for the basic investigations of the plasma confinement in a different magnetic configurations. Also, power deposition region has been investigated in experiments with modulated gyrotron heating power. In this work, the first plasma potential and total secondary current profiles measurements results are presented in a comparison with Langmuir probes data.

RESENT MEASUREMENTS IN TJ-II STELLARATOR BY HIBP DIAGNOSTIC OF THE ELECTRIC POTENTIAL EVOLUTION AND FLUCTUATION IN ECRH AND NBI PLASMAS

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Heavy Ion Beam Probe diagnostics is used in TJ-II stellarator has been upgraded for two point measurements to study directly with a good spatial (up to 1cm) and temporal (up to 2 μs) resolution the plasma electric potential and density, poloidal component of electric field Ep and to extract radial turbulent particle flux $\Gamma_r = \Gamma_{EpolxBtor} = \Gamma_{ExB,}$ for the first time in stellarators.

. Low density ($n_e = 0.3 \div 0.5 \times 10^{19}$ m^{-3}) ECRH plasma in TJ-II is characterized by positive plasma potential ($\varphi(0) = +600 \div +400$ V). At higher densities the minor area of the negative electric potential appears at the edge. This area increases with the density, finally makes potential fully negative. This tendency is affected by ECRH power and deposition area.

. Recent experiments with Li-coating and NBI heating have shown evidence for spontaneous L-H transition in the TJ-II stellarator occurring at a threshold value of the plasma density. NBI plasmas in L mode are characterized by negative electric potential in the full plasma column from the center to the edge. The absolute value of the central potential is of order – 300÷– 600 V. These observations are independent on the magnetic configuration. Density rise (particle confinement, energy confinement) is associated with the rise of the negative $E_{r;}$ suppression of the turbulence. This observation lies inside the paradigm of the turbulence suppression by E_r as a mechanism of confinement improvement.

At the spontaneous L-H transition, which happens in the purely NBI heated plasma the core electric potential becomes more negative at the outer half of the plasma column. At the back H-L transition the plasma potential recovers to its L value.

During the direct L-H transition edge and core fluctuations of local plasma potential and poloidal electric field Epol shows some reduction.

Alfven modes were observed by HIBP in a core and edge plasma. Correlation studies with reflectometry, magnetic probes and Langmuir probes can give an insight to spatial structure and properties of the Alfven modes.

. These modes were appeared as in the potential, so in the density. They are mainly manifested in the plasma density and are not much pronounced in the potential fluctuation. As a rule, L-H transition is accompanied by Alfven modes suppression.

Production and electron heating of over-dense plasmas by 2.45 GHz electron Bernstein waves on CHS

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Production and heating of over-dense plasmas by 2.45 GHz microwave system under very low field condition was performed and demonstrated on CHS. In this experiment, microwave systems were arranged to aim at taking place mode conversion of launched electron cyclotron wave effectively into electron Bernstein wave (EBW). One system (ECH#1) is launched nearly perpendicularly to the toroidal field, where FX-B scenario is expected. Another (ECH#2) is launched obliquely for O-X-B scenario. Power deposition profiles in produced over-dense plasmas were measured directly by using power modulation technique at various magnetic configurations where electron cyclotron resonance (ECR) and upper hybrid resonance (UHR) layers were scanned in space widely[1]. Figure 1 is radial profiles of electron temperature, electron density and electron pressure, the response in p_e to ECH#1 or ECH#2 source, the coherence and phase lag at B_{ax}/B_{res} = 50 % (B_{res} = 875 G). Electron density exceeds about 3 times O-mode cutoff density ($n_{co} \sim 7.5 \times 10^{16}$ m^{-3}). Power deposition profile corresponds to δp_e-profile and the peak position is in over-dense region. No obvious difference in the deposition region between ECH#1 and ECH#2 was observed. It is not clarified yet whether or not the abovementioned mode conversion scenario occurs effectively. Moreover, it is unclear how the incident waves are transmitted and absorbed. We have been investigating wave trajectories, power absorption mechanism and mode conversion by using a ray-tracing method developed for LHD plasmas[2]. Figure 2 shows examples of ray trajectories injected from ECH#1 and ECH#2 calculated as O-X-B scenario. In the presentation, we will show experimental results of 2.45 GHz EBW heating and numerical results by a ray-tracing method.

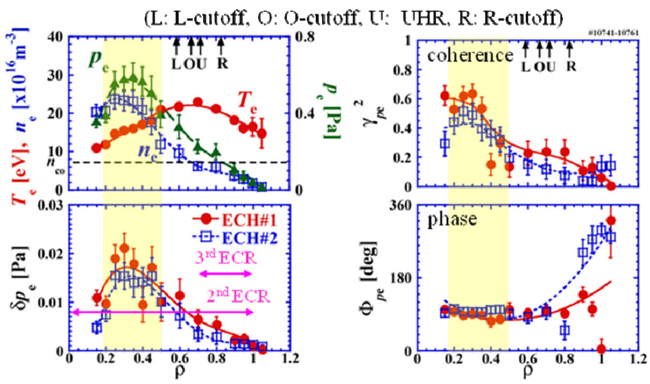

Fig.1 Radial profiles of electron temperature, T_e, electron density, n_e, electron pressure, p_e, the response in p_e to ECH#1 or ECH#2 source, δp_e, and the coherence, γ_{pe}^2, and phase difference between δp_e and the modulated power, Φ_{pe}, at B_{ax} = 437.5 G.

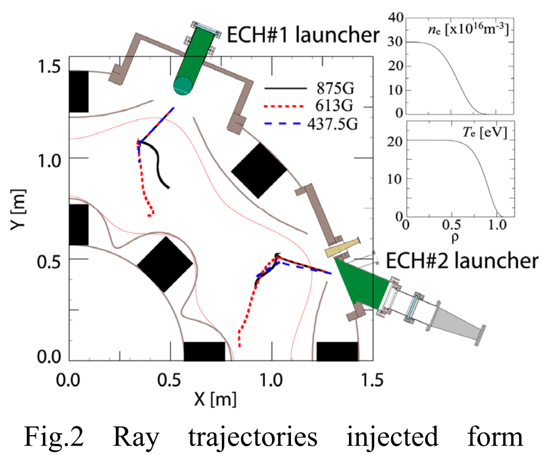

Fig.2 Ray trajectories injected form ECH#1 and ECH#2 launcher at B_{ax} = 875, 613 and 437.5 G

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Effects of gas-fueling control on plasma performance in Heliotron J

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This paper discusses the effects of the gas-fueling by the supersonic molecular beam injection (SMBI) technique on plasma performance in Heliotron J.

The gas fueling control is one of the most important factors to obtain a high density and good performance plasma from two aspects as well as the recycling control; (1) the profile control of the core plasma through the controlled penetration depth of neutral particles and (2) the reduction of neutral particles in the peripheral region. Supersonic molecular beam injection, which has been developed by L. Yao et al. [1, 2], is one method to obtain deeper penetration compared to the normal gas-puffing. Therefore, this technique is considered to be especially effective for a medium sized device.

Recently high-pressure SMBI is examined as a fueling method in Heliotron J [3]. The initial plasma in Heliotron J is produced by using the second harmonic X-mode ECH (70 GHz, < 0.45 MW). The hydrogen neutral beam (< 30 keV, < 0.7 MW/beam-line) is injected using two tangential beam-lines facing each other (BL-1 and BL-2). The SMBI system is installed on a horizontal port in Heliotron J. This system is originally introduced for the diagnostic purpose such as the gas-puff imaging measurement of edge plasma turbulence with a fast video camera [4]. By increasing the plenum gas pressure (≥ 1 MPa) of SMBI, fueling control is successfully applied to ECH/NBI plasma. Although the optimization of this method for the Heliotron J experiment is in progress, in a combination heating condition of ECH (~ 0.35 MW) and NBI (~ 0.6 MW), the stored energy reached ~ 4.5 kJ, which is about 50 % higher than the maximum one achieved so far under the normal gas-puff fueling condition [5] in Heliotron J. In addition, interesting time responses caused by the SMBI are observed, suggesting non-local transport phenomena: increase/decrease of T_e and its target density dependence for ECH plasma, two different types of propagation of perturbations in the radiation profile caused by the SMBI for ECH+NBI plasma.

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Electron density profile measurement in Heliotron J with a microwave AM reflectometer

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Measurement of electron density profile is an important issue for the understanding of plasma confinement. The goal of this study is to develop a microwave reflectometer for Heliotron J to investigate the particle transport. The basic design of the reflectometer is as follows; an amplitude modulation (AM) system [1] is adopted to reduce density fluctuation effects. The X-mode is selected as the propagation mode since even hollow density profile can be measured, which is typically observed in ECH plasmas in helical systems. By using the carrier microwaves of 33-56 GHz (Q-band), it is possible to obtain the density profile over almost the full range of plasma radius for low-density plasmas ($\bar{n}_e = 1.0\times10^{19}$ m^{-3}) and the density profile at edge region for high-density plasmas like edge transport barrier. The time resolution of the profile measurement with this system is less than 1 ms, which is shorter than the typical confinement time of Heliotron J plasma.

A pulse generator supplies a triangular-wave shaped voltage to a voltage controlled oscillator (VCO) of 8-14 GHz. The frequency band of 33-56GHz is generated by the VCO and x4 frequency multiplier. The output power from the multiplier is about 5 dBm. The microwaves are transmitted through 6 m oversized waveguides (Ka-band). After amplification, the waves are modulated in amplitude by using a PIN modulator, whose modulation frequency is 200 MHz. The launching and receiving antennas are located inside the vacuum vessel to remove the parasitic reflection. In the receiving system, a low-pass filter is assembled to suppress the effect of 70 GHz ECH. The reflected signal is detected by a diode detector with amplifier. A phase meter consists of frequency down-converters (2 MHz), limiting amplifiers, an amplitude detector and an I/Q phase detector.

The overall performance of the system has been tested by using a curved aluminum plate instead of plasma cutoff layer. Here the length of the transmission line is scanned by moving the plate position in the range of 40 cm at 1 mm intervals. The measurement results show that each microwave component works well and the measured phase shift is in agreement with theory. The density profile will be measured in the forthcoming experimental campaign.

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Study of ECCD Physics and Iota Profile Control in Heliotron J

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Non-inductive current has an important role on realization of high performance plasma and sustainment of steady state in toroidal fusion devices. In stellarator/heliotron (S/H) devices, electron cyclotron current drive (ECCD) is expected as an effective current drive scheme to tailor the rotational transform profile in order to suppress magnetic islands. In 70GHz 2nd harmonic ECCD experiments on Heliotron J, the EC driven current has been successfully separated from the bootstrap current by comparing the measured current in co- and counter-magnetic fields [1]. The experiment results show that the EC driven current flows in the direction determined by the Fisch-Boozer effect when the EC power is deposited on the top of magnetic field ripple, while it flows in the opposite direction determined by the Ohkawa effect when the EC power is deposited on the bottom of magnetic field ripple [2]. The normalized current drive efficiency, $\zeta=e^3 n_e I_{EC} R/\varepsilon_0^2 P_{EC} T_e$, is almost constant regardless of EC power at the density of n_e=0.5×10^{19} m^{-3}. This efficiency is found to be similar in Heliotron J, CHS and TJ-II [3]. The efficiency depends on the magnetic field configuration, indicating that the population of trapped electrons strongly affects the current drive efficiency. The poloidal field generated by localized EC driven current modifies the rotational transform. A simple calculation shows that the localized current of 5 kA generates negative shear or high positive shear in the core region of Heliotron J.

The injected beam was unfocused and the injection angle was fixed in the ECH/ECCD system used so far. The launching system is being upgraded by introducing a focusing mirror and a steerable flat mirror in order to localize the EC power and to control the power deposition position. The low power test results show that the available parallel refractive index ranges from -0.1 to 0.6, and the $1/e^2$ beam radius at magnetic axis is 3 cm, smaller than the minor radius, a~20 cm. We will conduct the ECCD experiment using this launching system in the forthcoming experimental campaign.

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On ECCD efficiency in toroidal plasmas with finite collisionality *

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The standard method for calculation of ECCD generated current in tokamaks and stellarators is the adjoint approach where the flux surface averaged current is given by a convolution of a quasilinear source term with a generalized Spitzer function (local current drive efficiency). This function is well studied for high collisionality regimes where it is equivalent to the classical Spitzer function, and in the long mean free path regime where a bounce averaging procedure can be used to reduce the dimensionality of the problem to 2D. In the general case of finite plasma collisionality, the kinetic problem to compute the local efficiency remains essentially 3D for tokamaks and 4D for stellarators. For this reason, this general case is not studied as well as cases in the asymptotical limits.

In this work, the drift kinetic equation solver NEO-2 which is based on the field line integration technique has been applied to compute the generalized Spitzer function in a tokamak with finite plasma collisionality. The resulting generalized Spitzer function has specific features which are pertinent to the finite plasma collisionality. They are absent in asymptotical regimes or in results drawn from interpolation between asymptotical limits. These features have the potential to improve the overall ECCD efficiency if one optimizes the microwave beam launch scenarii accordingly. Such results and their implications for ECCD in tokamaks and stellarators will be discussed in the report.

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Conceptual Design of an X-ray Imaging Crystal Spectrometer for the Large Helical Device

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This paper presents the conceptual design of an x-ray imaging crystal spectrometer for the Large Helical Device (LHD), which can provide radial profiles of the ion temperature with a spatial resolution of 1 cm and a time resolution of $\geq$ 10 ms. In addition to ion temperature profiles, it should also be possible to obtain profiles of the poloidal rotation, the electron temperature, and argon ion charge state distributions, which are of interest for impurity transport studies. This instrument could be installed on LHD in October 2010. A prototype of the proposed spectrometer was thoroughly tested on Alcator C-Mod, where it has made significant contributions to the experimental program.

First observations of partially neutralized and quasineutral plasmas in the Columbia Non-neutral Torus

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The Columbia Non-neutral Torus (CNT) is a stellarator designed to study pure electron, partially neutralized and positron-electron plasmas. To date, CNT usually operates with electron rich plasmas (with negligible ion density) [1]. The accumulation of ions alters the equilibrium of electron plasmas in CNT and a global instability has been observed when the ion fraction exceeds 10%. A characterization of this instability is presented in [2], analyzing its parameter dependence and spatial structure (non-resonant with rational surfaces). A segmented conducting mesh has been installed to impose the electrostatic boundary condition V=0 at the plasma edge. Results after the installation of the conducting boundary confirm the presence of an instability in electron rich plasmas with a finite ion content. A new set of experiments is currently underway studying plasmas of arbitrary degree of neutralization, ranging from pure electron to quasineutral plasmas. Basic observations show that the plasma potential decouples from emitter bias when we increase the degree of the neutralization of our plasmas. Partially neutralized plasmas are also characterized by multiple mode behavior with dominant frequencies between 20 and 200 kHz. When the plasma becomes quasineutral, it reverts to single mode behavior (2 - 18 kHz). The first results on partially neutralized and quasineutral plasmas in CNT will be presented.

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Confinement of Pure Electron Plasmas in the Columbia Non-neutral Torus

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The non-neutral plasmas studied in the Columbia Non-neutral Torus are uniquely different from quasi-neutral plasmas in similar magnetic geometries. Confinement is primarily limited by transport resulting from insulating probes inserted into the plasma and electron neutral collisions. Recently the best measured confinement time has been increased by over an order of magnitude to 323 ms. An improved conducting boundary is being designed and installed to minimize potential variation along magnetic surfaces and externally diagnose the plasma. Three stellarator configurations can be studied in CNT by varying the angle between coils. Progress toward a comparison of each configuration will be presented. Methods to create plasmas unperturbed by internal rods and diagnose the plasma at or outside the edge are described and initial results are presented.

Effect of Nonaxisymmetric Perturbation on Profile Formation

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A superdense core (SDC) plasma of more than ~ 1×10^{21} m^{-3} develops with the formation of the internal diffusion barrier (IDB) when a series of pellets is injected into the neutral beam heated (NBH) plasma in LHD [1]. From the wide range magnetic configuration study, it has been found that the SDC plasma can be obtained only in the outward shifted configuration where the stochastic layer surrounding the confinement region is very thick. During the IDB-SDC discharge, a large Shafranov shift due to the high central plasma pressure takes place, which strongly modifies the magnetic field structure. According to the HINT2 code which can deal with the three-dimensional equilibrium, the ergodization develops with the increase in the central beta value. In the stochastic region, it is expected to have different heat and particle transport properties from the region with perfectly nested flux surfaces. It is surely observed in the experiment that the region where the density and its radial gradient are low spreads outside the SDC.

In order to see the relation between the magnetic field structure and the profile formation, resonant magnetic perturbations (RMPs) are applied to actively modify the magnetic field structure. Because of some restrictions, RMPs with only m/n = 1/1 and 2/1 are available so far in LHD, where m and n are poloidal and toroidal mode numbers, respectively. In outward shifted configurations, the resonance for RMP with m/n = 1/1 is in the stochastic region, on the other hand, RMP with m/n = 2/1 is in the closed region.

RMPs were applied to the SDC plasma of which edge region is already stochastic. It was found in the experiment that the density pump out is enhanced with RMP. On the other hand, an increase in the electron temperature in the core region was also observed. So far it has not been clear if those phenomena mentioned above affect each other. Numerical calculation suggests that the central deposition of the NBH power is enhanced with the reduction of the electron density in the edge region where the NBH power is absorbed. Furthermore another observation concerning the healing of magnetic islands or stochasticity is a topic of interest.

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Non-ambipolar particle fluxes and neoclassical viscosity in quasi-symmetry

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Plasmas confined in a geometry with an ignorable, or 'symmetry' coordinate, have superior particle orbits over systems that are 3-dimensional. However, for systems with perfect symmetry, collisional transport theories do not predict the flow along the symmetry direction to lowest order. Having some influence on flows along the direction of symmetry is highly desirable. The introduction of weakly non-symmetric magnetic fields generates mirror and curvature forces with a component in the direction of symmetry that drives non-ambipolar radial particle drifts. In a fluid moment approach [1], symmetry breaking forces appear as a modification to the component of the parallel viscous stress tensor in the direction of symmetry. In the context of tokamaks, where the symmetry coordinate is the toroidal angle, this force is termed neoclassical toroidal viscosity [NTV] [2], and vanishes in the limit of perfect axisymmetry.

In a quasi-helically symmetric [QHS] device, there exists a dominant helical symmetry angle $\alpha_h \equiv \theta - M\zeta/N$, in which M, N are fixed integers such that $|B| = B_0\,(1-\varepsilon_h \cos M\alpha_h)$ along a field-line, with several much smaller helical 'sidebands.' The symmetry angle α_h is analogous to the poloidal direction in tokamaks, and equivalently to the toroidal direction in a tokamak, there exists a direction of near helical symmetry and thus least flow damping along $\vec{e}_{\zeta_h}$ such that $\vec{e}_{\zeta_h} \cdot \vec{\nabla}\alpha_h = 0$. Provided the departure from symmetry is small, i.e. $\delta B_{\text{eff}}/B_0 \ll \varepsilon_h$ where $\delta B_{\text{eff}}/B_0$ is the effective helical sideband strength, flow damping and thus flow evolution along ($\vec{e}_{\zeta_h}$) and 'cross' ($\vec{e}_{\alpha_h}$) the direction of symmetry in a flux surface decouple [3, 4]. In this case, the weaker flow damping in the symmetry direction $\vec{e}_{\zeta_h}$ driven by non-helically symmetric magnetic fields is of the form

$$\frac{\partial \Omega}{\partial t} = -\mu_{\parallel}\left(\frac{\delta B_{\text{eff}}}{B_0}\right)^2 (\Omega - \Omega_*)\,. \tag{1}$$

Here Ω is the rotation rate along the symmetry direction, $\mu_{\parallel}$ is the neoclassical viscous damping rate, and Ω_* is a diamagnetic-type 'offset' rotation, to which the non-helically symmetric magnetic fields damp the flow. The specifics of the damping rate and the rotation 'off-set' depend on the collisionality and Ω. Various banana-drift regimes have been derived for tokamaks and QHS stellarators [4]. In this context, a complete version of (1) describing all forces acting in the $\vec{e}_{\zeta_h}$ direction is equivalent to the ambipolarity condition. Steady state solutions of this rotation equation are equivalent to 'roots' of the ambipolarity condition in conventional stellarator theory, to determine the radial electric field profile. Steady-state 'rotation roots' of (1) will be presented for quasi-symmetric stellarators under the dominant influence of neoclassical flow damping for various banana-drift regimes. This research was supported by the U.S. Department of Energy under Grant Nos. DE-FG02-86ER53218, DE-FG02-92ER54139 and DE-FG02-99ER54546.

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Modeling and Measurement of Toroidal Currents in the HSX Stellarator

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A set of magnetic diagnostics, including Rogowski coils, diamagnetic loops and two poloidal 'belts' of 3-axis pick-up coils measure the magnetic field at several locations around HSX. The Rogowski coils and diamagnetic loops measure changes in the net toroidal current and toroidal flux, respectively. The two "belts" are separated by ~ 1/3 of a field period and measure the local magnetic field vector at 16 poloidal locations at the two toroidal angles. With the BOOTSJ [1] and VMEC [2] codes, a self-consistent calculation of the Pfirsch-Schluter (PS) and bootstrap currents is performed. The 3-D equilibrium reconstruction code V3FIT [3] is then used to calculate the expected response of the magnetic diagnostics. The sensitivity of the diagnostic set to features in both the PS and bootstrap current is explored. Because of the lack of toroidal curvature, the dipole PS current has a helical rotation and nearly reverses at the two toroidal locations. The bootstrap current is opposite in direction in HSX compared to that in a tokamak. This reduces the rotational transform (~1) but increases the effective transform (~3). Compared to a tokamak, the relative magnitude of each current is reduced by a factor of the effective transform (~3). The PS current reaches steady state quickly, whereas the bootstrap current rises throughout the discharge on a 50-500 ms timescale. In most cases, the toroidal current is still rising at the end of the discharge, reaching 0.4-0.6 kA. Several improvements in the modelling are now in progress. One is the comparison of the measured bootstrap current to the PENTA code [4] which includes the effects of parallel momentum conservation and finite electric field. Second is the comparison of the evolution of the net toroidal current (bootstrap+induced) to a 3-D model that relies on the calculation of the susceptance matrix [5]. Third is an analysis of the sensitivity of the magnetic diagnostic set and the ability of the V3FIT code to reconstruct the plasma pressure and current profiles based on the magnetic signals.

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Local effects of magnetic resonances in ECRH plasmas of the TJ-II Heliac

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One of the peculiarities of the Heliac design of stellarator magnetic fields is its wide magnetic configuration space. In the TJ-II one, rotational transform (ι) values ranging from 1 to 2 can be easily accessed in low –vacuum– magnetic shear conditions. Other properties linked to magnetic configuration, like magnetic well or enclosed plasma volume, can be explored as well. So far, most of the work has been dedicated to ι scans. These studies have been boosted with the commissioning of a system to perform dynamic magnetic configuration scans, where the offset of the ι -profile can be changed in a single (~300 ms) discharge at practically constant plasma volume and no ohmically induced currents [1]. In contrast with dynamic scans aided by the induction of ohmic currents, in these new scans the net plasma current behaves like in static configuration discharges. All of them, static and dynamic ι –scans, have proven that in the conditions of these plasmas (ECR heating ~500 kW, average densities $\leq 1.2 \times 10^{19}$ m^{-3}, ~1 keV central electron temperature) low order rational values of ι (magnetic resonances, for short) can be noticed throughout the plasma volume due to their effect on, for instance, electron density and temperature gradients. Therefore, the effect can be seen in quite different conditions of collisionality and magnetic shear. Interestingly, the experiments point to *increased* gradients around the location of the magnetic resonances. In addition, several diagnostics show that there is a differential rotation of the plasma corona where the resonance is present, which clearly shows a local modification of the radial electric field.

The results presented indicate that pure collisional and ExB shear dependent (e.g. electrostatic turbulence) contributions to transport must be considered together in presence of the magnetic resonances. As long as the latter do not destroy confinement, something proven here to be possible even in a low magnetic shear scenario, the resonances can be seen as volume-filling structures between well formed magnetic surfaces, Ψ. In a stellarator, where the different mobility of electrons and ions oblige the exploration of flux surfaces $\mathrm{d}\Psi / \mathrm{d}t = \mathbf{v}_\mathrm{d} \cdot \mathrm{grad}\,\Psi$ to be different among plasma species (here $\mathbf{v}_\mathrm{d}$ represents the drifts due to magnetic spatial variations), magnetic resonances are quite likely to provoke localized sheared electric drifts. Therefore, their importance as effective confinement knobs should not be dismissed. To help in quantifying the modification of collisional transport due to the magnetic resonances, we have started a program of calculations using a new drift-kinetic code based on the evolution of the electron and ion distribution functions in non-axisymmetric toroidal geometry –inluding TJ-II fields.

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Enhancement of Residual Zonal Flows in Helical Systems with Equilibrium Radial Electric Fields

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We have been investigating the interaction between zonal flows and turbulence in helical systems such as heliotrons and stellarators based on gyrokinetic theory and simulation [1-5]. In helical systems, the equilibrium radial electric field *Er* determined from the ambipolar particle flux condition generates the macroscopic ***E***×***B*** rotation. This ***E***×***B*** rotation, which is distinguished from the microscopic sheared ***E***×***B*** zonal flows, is expected to further enhance the residual zonal-flow level as pointed out by Mynick and Boozer [6] who used the action-angle formalism to treat poloidally closed ***E***×***B*** drift orbits of helical-ripple-trapped particles. We have taken account of the closed and unclosed ***E***×***B*** drift orbits, the latter of which can transit to the toroidally-trapped orbits with some probability, and derived the formulas for collisionless zonal-flow responses [5]. Consequently, the zonal-flow responses are predicted to be raised either by neoclassical optimization of the helical geometry lowering the radial drift or by strengthening *Er* to boost the poloidal rotation. In order to confirm the prediction, the time evolution of the zonal-flow potential is solved by using the gyrokinetic Vlasov (GKV) code, of which the simulation domain is extended from a flux tube to a poloidally-global region [4]. Figure 1 shows results for the cases of $M_p = 0$ ($E_r = 0$) and $M_p = 0.3$ where M_p represents the poloidal Mach number of the ***E***×***B*** drift velocity. Here, the model geometry of the inward-shifted LHD configuration is used. The enhancement of the residual zonal-flow level due to *Er* is clearly verified. We find that this enhancement due to *Er* is more evident for the inward-shifted case than for the standard LHD. The *Er* effect appears through *Mp*. When the magnitude of *Er* and the magnetic geometry are fixed, higher zonal-flow responses are obtained by using ions with a heavier mass, which increases *Mp*, and accordingly the resultant turbulent transport is expected to show a more favorable ion-mass dependence than the conventional gyro-Bohm scaling.

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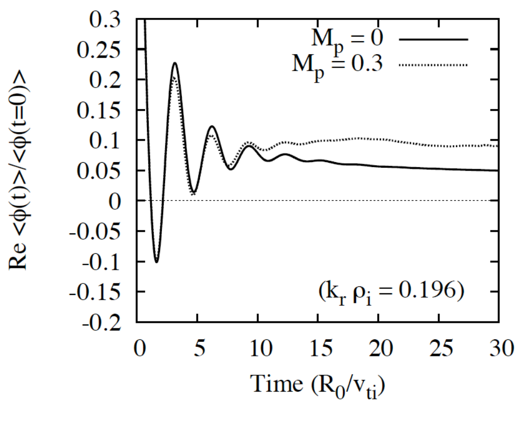

Fig.1 Time evolution of the zonal-flow potential

Tuesday

Oct. 13, 2009

08:30 – 09:15	PL01	H.Yamada	***High Beta Issues in a Helical System***
09:15 – 09:45	I09	W.A. Cooper	*Drift stabilization of ballooning modes in a high-$<\beta>$ LHD configuration*
09:45 – 10:15	I10	Y. Narushima	*Experimental study of effect of poloidal flow on stability of magnetic island in LHD&TJ-2*
10:15 – 10:45	I11	D. Pretty	*Results from an international MHD data mining collaboration*

Coffee

11:15 – 11:45	I12	A. Reiman	*3D Equilibrium with Stochastic Regions Having Finite Pressure Gradient*
11:45 – 12:15	I13	S.R. Hudson	*Cantori, chaotic coordinates and temperature gradients in chaotic fields*
12:15 – 12:45	I14	J.D. Hanson	*Three-dimensional Equilibrium Reconstruction: The V3FIT Code*

Lunch

13:45 – 15:40	P2-01	T. Akiyama	***Status of a stellarator/heliotron H-mode database***
13:45 – 15:40	P2-02	H. Funaba	*Data Servers for the International Stellarator/Heliotron Profile Database (ISHPDB)*
13:45 – 15:40	P2-03	H. Funaba	*Transport Analysis of Reactor-Relevant High-Beta Plasmas on LHD*
13:45 – 15:40	P2-04	E. Ascasibar	***Global energy confinement studies in TJ-II NBI plasmas***
13:45 – 15:40	P2-05	B. Zurro	***The level of non-thermal velocity fluctuations deduced from Doppler spectroscopy …***
13:45 – 15:40	P2-06	J.A. Baumgaertel	***Linear and nonlinear gyrokinetic studies of turbulence in stellarator geometry with GS2 …***
13:45 – 15:40	P2-07	R. Kleiber	***Global gyrokinetic simulations for stellarators***
13:45 – 15:40	P2-08	H.E. Mynick	*Geometry dependence of stellarator turbulence via GENE*
13:45 – 15:40	P2-09	G. Birkenmeier	***Geometrical Magnetic Field Effects on Turbulent Transport***
13:45 – 15:40	P2-10	C.D. Beidler	***Momentum-correction techniques for stellarators***
13:45 – 15:40	P2-11	H. Maaßberg	***Momentum Correction Technique for Neutral Beam Current Drive***
13:45 – 15:40	P2-12	N.B. Marushchenko	*Momentum Correction Techniques for ECCD*
13:45 – 15:40	P2-13	Yu. Turkin	***Impact of momentum correction on bootstrap current and ECCD in W7-X***
13:45 – 15:40	P2-14	Yu. Turkin	***Benchmark of neoclassical thermal transport matrix***
13:45 – 15:40	P2-15	A.S: Ware	***Neoclassical and anomalous flows in stellarators***
13:45 – 15:40	P2-16	S. Nishimura	*A* ***convergence study for the Laguerre expansion in the moment equation method for …***
13:45 – 15:40	P2-17	A. Matsuyama	***A Monte-Carlo-based calculation of neoclassical flows and viscosity for nonaxisymmetric…***
13:45 – 15:40	P2-18	A. Wakasa	***Development of neoclassical transport module for …TASK3D***
13:45 – 15:40	P2-19	S. Matsuoka	***Status of a stellarator/heliotron H-mode database***
13:45 – 15:40	P2-20	O. Bondarenko	***Influence of low order rational surfaces on the radial electric field of TJ-II ECH plasmas***
13:45 – 15:40	P2-21	Satake	***Calculation of neoclassical toroidal viscosity in toroidal plasmas with broken symmetry***
13:45 – 15:40	P2-22	R. Reimer	***Spectrally resolved Motional Stark-Effect polarimetry at ASDEX Upgrade….***
13:45 – 15:40	P2-23	K. Ida	***Ion internal transport barrier in Large Helical Device***
13.45 – 15:40	P2-24	M. Yokoyama	*Considerations from the viewpoints of neoclassical transport towards higher ion-..*

15:40 – 16:00	C05	J.M. Canik	*ELM triggering by non-axisymmetric magnetic fields in NSTX*
16:00 – 16:20	C06	J.-K. Park	*Ideal Perturbed Equilibria in Tokamaks and Control of External Magnetic Perturbations*
16:20 – 16:40	C07	D. Terranova	*Self-organized helical equilibria in the RFX-mod reversed field pinch*
16:40 – 17:00	C08	D. Carralero	*Radial and Parallel Transport Fast Camera Observation on LHD SOL Region*
17:00 – 17:20	C09	N. Nishino	*Peripheral turbulence measurement in Heliotron J using fast cameras*
17:20 – 17:40	C10	M. Otte	*Overdense Plasma Operation in WEGA Stellarator*
17:40 – 18:00	C11	T. Sunn Pedersen	*Overview of results from the CNT stellarator and future plans*

High Beta Issues in a Helical System

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Recent experiments in LHD have demonstrated reactor-relevant high beta plasmas. The high beta state with the volume averaged beta of 5 % has been maintained for longer than 100 times the energy confinement time without any hazardous instability. Accumulated data and knowledge is a firm basis for a reactor assessment and also cultivates a new horizon of plasma physics. This talk tries to sort out available experimental observation to clarify our degree of understanding of high-beta plasmas in a helical system. The scope is not limited to MHD stability and equilibrium and issues of transport are also discussed. For example, it is one of major achievements in LHD to provide the evidence that MHD instability in magnetic hill is benign. The linear MHD theory has been so successful for current-driven instability in tokamak that this physical model has been recognized as a golden rule in fusion research. Experimental observation in LHD addresses the review to validate the physical model for pressure driven instability. Also it should be noted that the MHD stability theory relies on the description of 3-D MHD equilibrium which does not necessarily guarantee the existence of nested flux surfaces. The effect of the Shafranov shift on transport is assessed with considering the working hypothesis that the suppression of neoclassical transport mitigates turbulent transport. The enhanced transport by the resistive interchange mode is also quantified with regard to the resistivity of plasma. Since the concept of the helical system is diversified, identification of commonalty and difference is discussed in comparison with other experiments such as W7-AS and even tokamaks.

Drift stabilization of ballooning modes in a high-$\langle\beta\rangle$ LHD configuration

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Ideal MHD yields at best inconclusive predictions about the stability of the LHD heliotron for $\langle\beta\rangle \geq 3\%$ [1]. We investigate the impact of the drift stabilization of ballooning modes for the inward shifted LHD configuration (vacuum magnetic axis $R_0 \sim 3.5m$). The background equilibrium is considered anisotropic in which the neutral beam ions contribute about a 1/4 fraction of the total diamagnetic beta, $\langle\beta_{dia}\rangle$. A kinetic ballooning mode equation obtained from the linearized gyrokinetic equation [2] is expanded assuming that the hot particle drifts are much larger than the mode frequency [3] to obtain a drift-corrected ballooning model. The fast particle pressure gradients contribute weakly to both the instability drive and the diamagnetic drift stabilization (which is dominated by the thermal ion diamagnetic drifts). At $\langle\beta_{dia}\rangle \sim 4.8\%$, the thermal pressure gradients drive ballooning modes in a broad region encompassing the outer $60 - 90\%$ of the plasma volume. The diamagnetic drift corrections (mainly from thermal ions) stabilize these modes. The energetic ion diamagnetic drifts play a role only for low wave number $k_\alpha < 10$. We have verified that the fast particle drift ordering imposed is amply satisfied for on-axis hot particle to thermal ion density $\sim 1\%$ at high $\langle\beta_{dia}\rangle$.

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Experimental study of effect of poloidal flow on stability of magnetic island in LHD and TJ-II

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The dynamics of magnetic island in the Large Helical Device (LHD) have been studied [1] in which a static magnetic island with m/n = 1/1 (here, m/n is the poloidal/toroidal Fourier mode number) is produced by perturbation coils. In the steady state condition, the stability of the magnetic island depends on beta (β) and collisionality (ν_h*) as shown in Fig.1. The magnetic island grows in the lower-β and higher- ν_h* regime (closed circles) while it disappears in the higher- β and lower- ν_h* regime (open circles). The mechanisms of stabilization of the magnetic island could not be explained by the effects of Bootstrap current and/or Pfirsch-Schlüter current [1]. Another effect of the poloidal flow should be considered because many studies have reported that the poloidal flow affects the magnetic island via the drag of magnetic island [2] and/or the ion-polarization current effect etc. [3-4]. In the LHD plasmas, the direction of the radial electric field, E_r, is thought to be a key parameter for understanding the stability of the magnetic island by the ion-polarization current effect [5]. When the plasma is in the electron-root ($E_r > 0$), the effect is stabilizing, which is consistent with the experimental observation of the CERC plasmas in TJ-II [6]. Positive E_r can be created in the neighbourhood of magnetic resonances, which will be able to heal them under certain conditions. Furthermore, in the TJ-II experiment of the dynamic ι scan, a positive spike of E_r from the negative value around the magnetic resonance has been observed by a Doppler reflectometer diagnostic. On the other hand, in the case of an ion-root ($E_r < 0$), the relation between the magnitudes of the $\boldsymbol{E}\times\boldsymbol{B}$ drift ($\omega_{\boldsymbol{E}\times\boldsymbol{B}}$) and ion diamagnetic drift (ω_{*pi}) is important. The calculated radial electric field [7] in the case that the magnetic island is stabilized shows the ion-root ($E_r < 0$) in the whole region of the plasma. The angular velocity of the $\omega_{\boldsymbol{E}\times\boldsymbol{B}}$ and ω_{*pi} assuming n_i (p_i) = n_e (p_e) at $\iota/2\pi = 1$ are $\omega_{\boldsymbol{E}\times\boldsymbol{B}} = -5.1\times10^3$ [rad/s] and $\omega_{*pi} = 4.0\times10^3$ [rad/s], respectively. The relationship of $|\omega_{\boldsymbol{E}\times\boldsymbol{B}}| > |\omega_{*pi}|$ indicates the island is stabilized. Further detailed analyses about poloidal flow will provide useful information for understanding the behavior of the magnetic island in TJ-II and LHD.

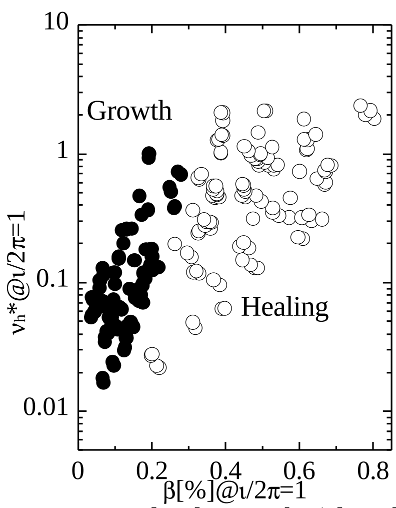

Fig.1 Island growth (closed circles) and healing (open circles) in β - ν_h^* space [1].

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Results from an international MHD data mining collaboration

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Data mining techniques have been successfully applied to MHD data on H-1, TJ-II and Heliotron-J, and are being implemented on LHD and W-7AS data. The motivation for automated mining of fusion databases is to extract physically-interesting, previously unknown information from existing data. In general, the database of a fusion device will be sparse, with some diagnostics in operation only for specific experimental campaigns; also, changes in diagnostic states (location, gain, filters, etc.) greatly complicate the data pre-processing required for unsupervised data mining procedures. Therefore, we focus initially on timeseries data from Mirnov coil arrays, which, in general, are always active and have fixed locations. Furthermore, restricting the automated process to the Mirnov phase information means we are not concerned with the signal amplitudes (i.e. variation in gain settings from shot to shot). Using unsupervised classification (clustering) of data produced by Fourier and SVD analysis, we extract different classes of Mirnov fluctuations as defined by the set of phase differences between neighbouring coils. From these classifications, we investigate relations between the fluctuations and other properties (e.g. density, B) using more deliberate, supervised methods. We present results from data mining of more than 10,000 shots from H-1, TJ-II and Heliotron J, showing a range of Alfvénic and non-Alfvénic modes, many with well-defined poloidal mode structure and clear relation to heating configuration and plasma geometry. In the case of H-1, the dispersion relations for several of these modes have been examined due to the available high resolution in rotational transform. Examples of use of this relation to provide information about rotational transform are given. We also discuss possible application of the cluster technique to preliminary mode identification as data is being acquired, and some initial work on application of image processing techniques to MHD spectrogram analysis. We report on the status of the LHD and W7-AS data mining projects, and also discuss progress towards incorporation of data from this process into a stellarator MHD documentation database, proposed last year at the 4th Coordinated Working Group Meeting (CWGM).

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3D Equilibrium with Stochastic Regions Having Finite Pressure Gradient*

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We have examined the nature of plasma equilibria in a field with chaotic field line trajectories. Equilibrium calculations for high beta shots in the W7AS and LHD stellarators have indicated the formation of a large region of stochastic field lines at the plasma edge, with a nonzero pressure gradient in that region. For diverted tokamaks, there has been some success in suppressing edge localized modes (ELMs) by the imposition of nonaxisymmetric fields near the plasma edge, and the possible role of the stochastic layer produced near the diverter separatrix is a subject of current research. More generally, whenever field line stochasticity is invoked as a possible contributor to anomalous transport, the nature of the underlying plasma equilibrium is at issue. We recast the equilibrium equations in a form where the stochasticity enters only through the magnetic differential equations along the chaotic field line trajectories that determine the Pfirsch-Schlüter current. We take advantage of a similarity in form between (i) the magnetic differential equation that determines the equilibrium Pfirsch-Schlüter currents (and a similar equation for the variation of the pressure along the field), (ii) the Liouville equation for magnetic field lines, and (iii) nonlinear equations for turbulent plasmas, such as the Vlasov or drift-kinetic equations, to apply mathematical methods of turbulence theory to the magnetic differential equations. This is not straightforward because the solutions of the Liouville equation and the equations for turbulent plasmas make use of causality, which cannot be invoked in solving for the equilibrium, and because of the need to impose periodicity in the toroidal and poloidal directions on the equilibrium solution. This work provides a firmer foundation for calculations using the PIES code to study high beta shots in W7AS, and for future calculations of equilibria in magnetic fields having stochastic field lines.

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Cantori, chaotic coordinates and temperature gradients in chaotic fields

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TOPIC :

Toroidal magnetic field line flow is a $1\frac{1}{2}$ dimensional Hamiltonian system and, in the absence of symmetry, such systems are generally chaotic. The existence of local regions of irregular trajectories has profound implications for a variety of problems in plasma confinement: here we consider anisotropic heat transport. The study of heat transport in chaotic fields has a long history [Rechester & Rosenbluth, PRL (1978)], but conventional approaches effectively treat chaotic fields as random, and thus overlook some important properties of chaotic fields, namely the hierachy of invariant dynamics comprised of regular, irrational trajectories. The invariant irrational sets, known as cantori, that persist after the destruction of the KAM surfaces can form effective partial barriers to anisotropic transport.

We demonstrate using a model of heat transport with separate parallel and perpendicular thermal diffusion coefficients, $\kappa_\parallel$ and $\kappa_\perp$. For fusion plasmas the ratio $\kappa_\parallel/\kappa_\perp$ may exceed 10^{10}, and the temperature adapts to the fractal structure of the magnetic field. This paper will show that temperature gradients coincide with the cantori.

We develop *chaotic*-magnetic coordinates [Hudson & Breslau, PRL (2008)]: coordinates adapted to the invariant structures of the field line flow. By adapting the coordinate surfaces to the partial barriers formed by the cantori, the coordinate surfaces coincide with isotherms. The temperature written in chaotic coordinates is well approximated by $T = T(s)$, where s is a radial coordinate, and an expression for the temperature gradient is derived:

$$\frac{dT}{ds} = \frac{c}{\kappa_\parallel \varphi + \kappa_\perp G}, \tag{1}$$

where $\varphi = \int\int d\theta d\phi \sqrt{g} B_n^2$ is the squared field-line flux across a coordinate surface, and $G = \int\int d\theta d\phi \sqrt{g} g^{ss}$, is an averaged metric quantity

Three-dimensional Equilibrium Reconstruction: The V3FIT Code

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Equilibrium reconstruction – using experimental data to determine MHD equilibrium properties of a confined plasma – has greatly improved the operation and interpretation of axisymmetric plasmas [1]. As modern stellarators and heliotrons achieve lower aspect ratios, larger plasma betas, and larger bootstrap currents, the deviation of equilibrium magnetic fields from vacuum configurations increases. Equilibrium reconstruction of the three-dimensional MHD equilibrium of stellarators and heliotrons will become more important to the accurate interpretation of experiments.

V3FIT is a three dimensional equilibrium reconstruction code written in Fortran 95, designed to be modular, flexible, extensible and fast [2]. Presently it uses VMEC [3] as the equilibrium solver. V3FIT minimizes a function

$$g^2(\mathbf{p}) \equiv \sum_i \left(\frac{S_i^o(\mathbf{d},\mathbf{p}) - S_i^m(\mathbf{p})}{\sigma_i} \right)^2 \tag{1}$$

where the **p** are parameters of the equilibrium (such as the parameters characterizing the current and pressure profiles), the **d** are experimental data from a particular time, the S^o (S^m) are the experimentally observed (model computed) signals, and the σ are variances in the signals. Currently, V3FIT uses magnetic diagnostics and geometrical limiters as signals.

V3FIT performs well on stellarator reconstructions using simulated experimental data. The axisymmetric equilibrium computation capabilities of VMEC and the magnetic signal calculations of V3FIT have been carefully benchmarked against the EFIT code. V3FIT performs comparably to EFIT on axisymmetric reconstruction using actual experimental data from the DIII-D tokamak. V3FIT's algorithm performs as expected in the presence of simulated experimental error and in ignoring singular directions in parameter space. The algorithm is capable of robustly reconstructing equilibria from far away in parameter space.

Current efforts at improving V3FIT include: incorporating its use into the routine operation of the Compact Toroidal Hybrid (CTH) experiment at Auburn, improving the description of geometrical limiters in the reconstruction, and adding soft X-ray diagnostics as signals.

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Status of a stellarator/heliotron H-mode database

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Confinement improvements with the edge transport barrier (ETB) like the H-mode in tokamaks are observed in several helical devices (CHS[1], Heliotron-J[2], LHD[3], TJ-II[4], W7-AS[5]). One of characteristics of helical devices is a wide variety of magnetic configurations: profiles of the rotational transform, island structures, the ripple, the aspect ratio, the edge stochasticity, and so on. In addition, the device size is also varied. There are large devices whose parameters are comparable to large tokamaks and small devices which have sophisticated diagnostics and tools (biasing, ergodization coils, etc.) to prove assumptions. Hence, comparative research among these devices is a powerful situation to understand the physics of the H-mode phenomena. Since the H-mode in tokamak is a standard operation in ITER, the study of the phenomena and the estimation of the required conditions such as the heating threshold power and the electron density have been energetically carried in the tokamak community. The understanding though the above study of the H-mode in helical devices will be fruitful information even for tokamaks. In addition, this will lead to the general understanding of confinement improvement modes in torus plasmas.

'H-mode and ELMs' session kicked off at the 5th Coordinator Working Group Meeting (CWGM) held at Stuttgart. To clarify differences and/or similarities of the different types of ETBs, we started to create database of the H-modes in each device. It will include set of profiles and global data prior to the transition and ones in the fully developed H-mode. Since edge parameters are considered to be strictly related to the H-mode transition, the detailed edge profiles and mapping of edge parameters at a certain edge position will be also included. Normalized parameters like collisionality derived from above data will be useful for inter-machine comparison.

At the ISHW, the present status of this database (registered data, data structures, and so on) will be reported.

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Data Servers for the International Stellarator/Heliotron Profile Database (ISHPDB)

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The WWW servers for the International Stellarator/Heliotron Profile Database (ISHPDB) [1] began to operate in order to register and open to public the profile data of the plasmas of severral stellarator/heliotron devices. The ISHPDB activity is a joint collaboration of the stellarator/heliotron community. The servers are located at the Max-Planck Institut für Plasmaphysik (IPP site: http://www.ipp.mpg.de/ISS/) and the National Institute for Fusion Science (NIFS site: http://ishpdb.nifs.ac.jp/). The registered data which are opened to the public are in principle already published data. The first step of ISHPDB is the collection of UFILEs on the WWW servers. UFILE is the file format of the ITER profile database [2]. The UFILEs are stored under the directories with the names of each device, which are 'LHD', 'W7-AS', 'TJ-II' at present. The lists of the physics themes indicate links to the associated UFILEs. The present physics themes are as follows; 'CERC' (Core Electron Root Confinement) [3], 'MHD/High-Beta', 'Iota/Shear', 'High-Performance', 'H-mode/ELM', 'Particle/Impurity', 'Standard-Plasmas'. One UFILE can belong to more than one physics theme. For example, one plasma in the 'H-mode/ELM' theme may be included in the 'High-Performance' theme which is oriented to the high $n\tau T$ plasmas. In order to compare the data of several devices which have different magnetic configurations, it is important that the definitions with respect to the magnetic configuration should be clearly documented. The directories for the magnetic configurations are prepared under each device directory. In the UFILEs for ISHPDB, a new section for the information of the data, which is named 'HEAD_COMMENT', is attached. This section has some pairs of a keyword and contents. The keywords are 'physics topics', 'author', 'upDate', 'source', and so on. Figures 1 and 2 show examples of the CERC pages, which contain several links to the UFILEs of the CERC data on the IPP server and the NIFS server, respectively. They contain the same data and figures. The 'MHD/High-Beta' and 'High-Performance' data are also registered and the number of the registered data will be increased.

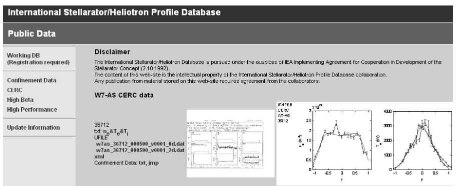

Fig. 1. A CERC page in the IPP server (http://www.ipp.mpg.de/ISS/).

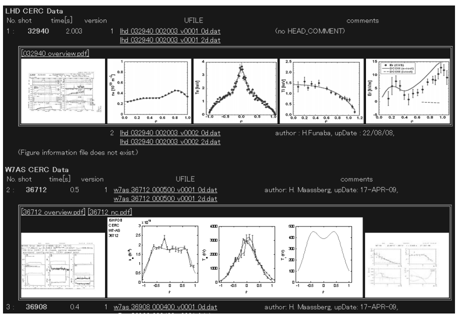

Fig. 2. A CERC page in the NIFS server (http://ishpdb.nifs.ac.jp/)

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Transport Analysis of Reactor-Relevant High-Beta Plasmas on LHD

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The dependence of the global confinement and the local transport of high-beta plasmas on the magnetic configuration or the volume averaged beta $\langle\beta\rangle$ has been studied for the LHD plasmas. For the plasmas of $\gamma = 1.254$, where γ is the pitch parameter of the helical coils, the effects of beta and magnetic configuration are evaluated separately in Ref. [1]. In order to produce high-beta plasmas, the optimization of the magnetic configuration has been made. The parameter γ is one of the most important parameters to obtain high-beta plasmas on LHD. Since the high-beta plasmas of more than $5\,\%$ were obtained in the magnetic configuration with $\gamma = 1.20$, the same method of analysis is applied for $\gamma = 1.20$ plasmas. The γ value is related with the aspect ratio A_{p}. In the configurations with the major radius of the preset magnetic axis position $R_{\mathrm{ax}}^{\mathrm{vac}} = 3.60\,\mathrm{m}$, $\gamma = 1.254$ and 1.20 correspond to $A_{\mathrm{p}} = 5.8$ and 6.6, respectively. In the $R_{\mathrm{ax}}^{\mathrm{vac}} = 3.60\,\mathrm{m}$ and $\gamma = 1.20$ configuration, the shift of the geomrtric center of the magnetic flux surface $R_{\mathrm{geo}}(\rho)$ at $\rho = 0.9$ is small in the region of $\langle\beta\rangle < 3\,\%$, while the shift in $\langle\beta\rangle > 3\,\%$ becomes large. With respect to the global confinement, the ratio of $\tau_{\mathrm{E}}^{\mathrm{exp}}/\tau_{\mathrm{E}}^{\mathrm{ISS04}}$ value, where $\tau_{\mathrm{E}}^{\mathrm{exp}}$ and $\tau_{\mathrm{E}}^{\mathrm{ISS04}}$ are the energy confinement time evaluated from the experiments and the ISS04 scaling [2], respectively, shows degradation at $\langle\beta\rangle < 2\,\%$. However, in the range of $\langle\beta\rangle > 2\,\%$, degradation seems to be small. The relations between $R_{\mathrm{geo}}(\rho)$ and the ratio $\chi^{\mathrm{eff}}/\chi^{\mathrm{ISS04}}$, where χ^{eff} is the experimental local transport coefficient and χ^{ISS04} is a transport coefficient which has the same non-dimensional parameter dependence as the ISS04 scaling, at $\rho = 0.5$ and 0.9 are shown in Figure 1 (a) and (b), respectively. From the comparison of the dependence in the high-beta plasmas (●) and in the low beta plasmas with three different magnetic configurations (△ or ○), the degradation of $\chi^{\mathrm{eff}}/\chi^{\mathrm{ISS04}}$ is almost same as the configuration effect at $\rho = 0.5$. On the other hand, at $\rho = 0.9$, the degradation of $\chi^{\mathrm{eff}}/\chi^{\mathrm{ISS04}}$ seems to be larger than the degradation by the configuration effect in $\langle\beta\rangle < 2\,\%$. The degradation in the high beta region of $\langle\beta\rangle > 2\,\%$ seems to be small in the case of $\gamma = 1.20$.

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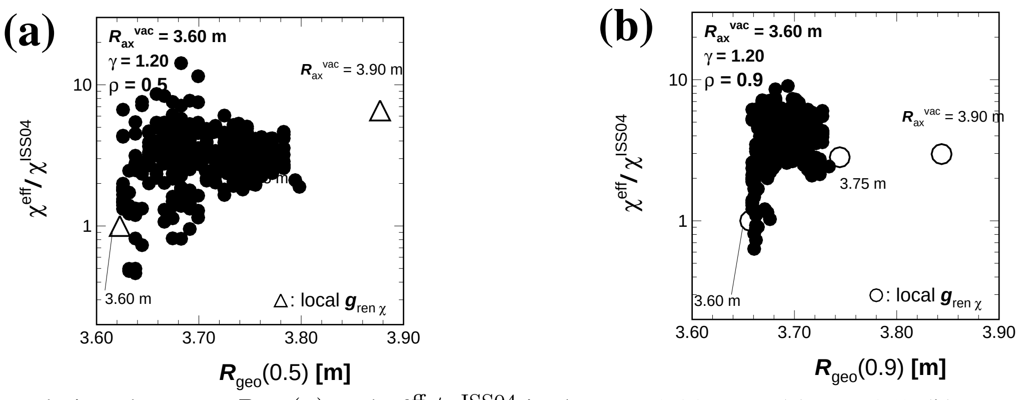

Fig. 1. Relations between $R_{\mathrm{geo}}(\rho)$ and $\chi^{\mathrm{eff}}/\chi^{\mathrm{ISS04}}$ in the $\gamma = 1.20$ case, (a) $\rho = 0.5$, (b) $\rho = 0.9$.

Global energy confinement studies in TJ-II NBI plasmas

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Earlier experimental studies in low density ($0.4\times10^{19}m^{-3} \le \bar{n}_e \le 1.2\times10^{19}m^{-3}$) ECH, boron-coated wall plasmas of the TJ-II stellarator (B ≈ 1 T, R = 1.5 m, <a> ≤ 0.22 m, $0.9 < \iota/2\pi(0) < 2.2$) yielded dependencies of the global energy confinement on operational parameters (plasma radius, heating power, density, rotational transform) in rough agreement [1] with the ones obtained from stellarator inter-machine scaling studies, like ISS04 [2]. Subsequent local transport studies extracted dependencies of the thermal diffusivity on the mentioned parameters, which confirmed and qualified the results of the global 0-dimensional study [3].

In the last year, operation under lithium-coated wall has given access to a new operational regime with long-lasting density control and improved confinement with NBI heating [4]. The present analysis concentrates on pure NBI plasmas (two injectors -co and counter-, beam energy ≤ 32 kV, current ≤ 60 A), with line average density values in the range $2\times10^{19}m^{-3} \le \bar{n}_e \le 6\times10^{19}m^{-3}$. The total estimated absorbed heating power (Fafner II Montecarlo code [5]) ranges from 125 to 750 kW. Regression analyses of the energy confinement time (up to 14 ms) on density and heating power yield a scaling law indicating substantially better global confinement than ECH plasmas. The confinement dependencies found (power exponent -0.8 and density exponent 0.4) indicate slightly stronger degradation with power and slightly weaker density improvement than ISS04, although the precise values of the exponents are uncertain due to the moderate correlation (0.6) between both magnitudes. Adequate coupling of the heating beam to the target plasma has been reached so far only for a limited range of (large) plasma radius and (medium) iota values. Therefore, these two magnitudes present a high co-linearity that precludes extracting accurate dependencies.

The still limited H mode database in TJ-II allows stating that there is a density threshold to access H mode, which depends on power level and magnetic configuration. The minimum required density measured is $2\times10^{19}m^{-3}$ at low heating power (200 kW absorbed). The strongest -although transient- H modes are obtained with both injectors (absorbed power around 700 kW) showing confinement improvement factors close to 60% over the L mode.

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The level of non-thermal velocity fluctuations deduced from Doppler spectroscopy and its role on TJ-II confinement

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The confinement scaling laws in stellarators depict general trends which need to be understood from basic mechanism accounting for them. The study of velocity fluctuations levels in the core of TJ-II intends to contribute to clarify these mechanisms. These fluctuations are elusive to standard techniques, and the limitations of the present approach are compensated by its uniqueness and by the use of high-resolution spectral techniques available for other purposes. The goal of this investigation is to study, in the line of previous works [1] and [2], the level of velocity fluctuations in different scenarios of the TJ-II stellarator. The method followed consist in measuring with spatial resolution, and time evolution for selected chords, the Doppler temperature of protons and C V, such as it has been explained in former publications [3-5]. The level of turbulent velocities in the plasma has been deduced from the difference observed between the apparent temperature of both species, following the method previously presented [1] and borrowed from astrophysics. The study of this difference, as a function of plasma density and injected power, provides a way to explore if this turbulence plays any role in the confinement of the hot TJ-II plasma.

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Linear and nonlinear gyrokinetic studies of turbulence in stellarator geometry with the GS2 code [1]

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The GS2 gyrokinetic code is being used to study microinstabilities and turbulence in flux-tubes in non-axisymmetric geometries, including stellarators. Stellarators have a number of interesting features, such as natural negative magnetic shear and a large number of shaping parameters, which offer possibilities for reducing microturbulence and improving performance. GS2 traditionally uses numerical equilibria for these studies, but recently an analytical stellarator equilibrium model [2] has been implemented for benchmarking with the GKV and GENE gyrokinetic codes. In addition to studying ITG modes, we will present results on kinetic ballooning mode growth rates and instability thresholds, using both the analytical model and numerical stellarator equilibria from recent design studies. Nonlinear results will also be discussed.

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Global gyrokinetic simulations for stellarators

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The code EUTERPE solves the gyrokinetic equation globally for fully three-dimensional equilibria using a particle-in-cell method. It has been extended recently to include electromagnetic perturbations, nonlinear terms and up to three kinetic species so that it allows the simulation of kinetic electrons and/or fast particles or impurities, respectively.
Thus it is possible to study the influence of kinetic electrons on the ITG instability in stellarators. Especially its influence on linear ITG modes and TEM in LHD will be investigated. A global code is necessary to investigate the influence of a neoclassical radial electric field on GAMs and residual ZF levels. We present linear simulations of the initial value problem for LHD including a constant radial electric field of varying size.

Geometry dependence of stellarator turbulence via GENE

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Using the stellarator gyrokinetic (gk) code GENE[1,2], we have computed the microturbulence & transport for a family of stellarators of widely-ranging designs, to understand the geometry-dependence of the turbulence. Existing nonlinear gk codes for stellarators are flux tube codes, giving a 1D picture of the turbulence along a particular field line. We have used a set of such field lines on a flux surface to construct the 2D structure of the turbulence over that flux-surface. We relate this structure to relevant geometric quantities, including the curvature, local shear, and effective potential $V_{ef}(\theta)$ in the Schroedinger-like equation governing linear drift modes, finding, for example, that the amplitude of ITG turbulence peaks in regions with the worst curvature, which occurs on the outboard side at the "corners" of the device, and that the ballooning of the turbulence is toroidal (toward $\theta=0$) in character, even for devices with dominant ripple, like QI/QOs or QHs.

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Geometrical Magnetic Field Effects on Turbulent Transport

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The three-dimensional structure of magnetic configurations of stellarators strongly affects the properties of the plasma dynamics. In particular, the magnetic field geometry has an effect on the formation and characteristics of plasma turbulence. In low-temperature plasmas in the torsatron TJ-K, the turbulent fluctuations can be measured by means of multi-probe arrangements in the whole confinement region. Experimental data of fluctuation amplitudes, cross-phases and turbulent transport is compared with the relevant parameters of the magnetic field geometry as magnetic curvature and magnetic shear. The general properties of the turbulence in TJ-K agree with drift waves. However, the influence of the magnetic configuration has been found in the cross-phase, which is sensitive to curvature effects. Measurements on a flux surface in the poloidal cross-section show increased fluctuation amplitudes and maximum transport in the region of bad curvature.

Momentum-Correction Techniques for Stellarators

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A neoclassical description of the transport processes in stellarators most commonly begins from solutions of the linearized *mono-energetic* drift kinetic equation in two spatial variables (typically poloidal and toroidal angle coordinates) and one velocity-space variable (e.g. the pitch-angle coordinate); the flux-surface label and the scalar velocity appear in this equation only as parameters. All relevant information from a solution may be condensed into just three mono-energetic transport coefficients allowing efficient storage of the results in a database for later use in transport calculations. The principal shortcoming of such an approach arises due to the lack of momentum conservation in the original solution of the kinetic equation which simplified the linearized collision operator by assuming only pitch-angle scattering. The importance of parallel momentum conservation is well known in calculations of the electric conductivity [1] and is commonly treated by introducing the so-called Spitzer function. A generalization of this approach has been developed with exact collisional and collisionless limits which makes use of the mono-energetic results at finite collisionality to model the “effective” fraction of trapped particles; a similar formalism can be employed to determine the bootstrap current. Results obtained in this manner have been successfully benchmarked with a second momentum correction technique, obtained by generalizing the method of [2] to an arbitrary number of moment equations, which can be shown to satisfy intrinsic ambipolarity of the particle fluxes for axisymmetric tokamaks at all orders of the expansion employed. This approach is very attractive for transport calculations since parallel momentum conservation is recovered by solving a small system of linear equations in which the parallel flows of all particle species are properly coupled by means of expressions which are, again, easily determined from the mono-energetic transport coefficients.

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Momentum Correction Technique for Neutral Beam Current Drive

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Neutral Beam Current Drive (NBCD) is traditionally estimated by the 1st Legendre harmonic of the slowing-down distribution function with a simplified collision term (Lorentz form) without momentum conservation and by the counter-acting electron Ohkawa current. The electron contribution is obtained from a Spitzer problem where momentum conservation is taken into account; see [1] for the collisional and [2] for the collisionless limit. This approach is generalised for arbitrary collisionalities both for tokamaks and stellarators using the parallel viscosity determined from the solution of the mono-energetic drift-kinetic equation; see [3]. Precalculated databases of all three mono-energetic transport coefficients from the DKES code [4,5] offer a very efficient basis for all momentum correction techniques.
In addition, the generalised Spitzer problem with the full linearised collision operator for all plasma species is solved. Here, the 1st Legendre harmonic of the beam ions, i.e. the slowing-down distribution estimated with the simplified Lorentz collision term, is used to calculate the 1st order Rosenbluth potentials which drive additional parallel flows (currents) of all species. This technique guarantees the overall momentum conservation in estimating the NBCD.
The impact of finite collisionalities and the full momentum conservation is shown for the reference ITER scenario 4 as well as for the W7-X scenarios with p-NBI and n-NBI.

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Momentum Correction Techniques for ECCD

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In electron cyclotron ray- and beam-tracing codes, the quasi-linear diffusion term is computed which is used in the adjoint approach for calculating the ECCD profile. For ray-tracing calculations, however, the solution of the DKE in the 4D-phase space for stellarators (3D for tokamaks) is much too time-consuming and fast approximations are needed. In the most common version of the adjoint approach, only the 1st Legendre harmonic of the solution of the linearised drift-kinetic equation (DKE) with a parallel momentum conserving collision operator is used being equivalent to a generalised Spitzer function.

The simple-minded *high-speed-limit* approach violates momentum conservation and can lead to a significant underestimation of the ECCD. Rather accurate approximations of the Spitzer function exist only for the collisional [1] and the collisionless [2] limit. Based on precalculated databases of mono-energetic transport coefficients, the Spitzer function is obtained for arbitrary collisionalities both for tokamaks and stellarators. Furthermore, a weakly relativistic approach [3] based on the variational principle given in [1] is benchmarked with the fully relativistic solution [4] in the collisionless limit.

The ECCD source function (i.e. the quasi-linear diffusion term), is highly localised in 4D-phase space. The collisionless solution can be extended to very small, but finite collisionalities. Except in the close vicinity of the maximum of B, the current diffuses from the passing particle region into a narrow sheath of barely trapped particles. This feature is approximated in a heuristic model which is currently undergoing benchmarking with the NEO-2 code for tokamak configurations [5].

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Impact of momentum correction on bootstrap current and ECCD in W7-X

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The minimization of the bootstrap current is one of the optimization criteria for the stellarator W7-X [1]. The bootstrap current alters the magnetic configuration, affects the rotational transform, and changes the spatial position of islands beyond the last closed magnetic surface where the divertor is located. The plasma current should be controlled within a range of about 10 kA in order to keep the shift of the island structure near the divertor plates within 3 cm. W7-X is not equipped with an Ohmic transformer, so the only sources of external current for control needs are electron cyclotron current drive (ECCD) and/or neutral beam current drive (NBCD). To this end, the development of realistic models to calculate bootstrap and driven currents represents important tasks for the W7-X experiment and for other fusion devices as well. The improved kinetic model by Taguchi [2] with momentum conservation in the collision operator and generalized formulation for the trapped-particle fraction has been included in the ray tracing code TRAVIS [3] and the 1D transport code [4]. Predictive transport modeling of ECR heated plasma in W7-X has been done using these improved models with focus on levels of the bootstrap and ECRH driven currents. In simulations the ion and electron energy balance equations with neoclassical fluxes are solved self-consistently with the equation for the ambipolar radial electric field. Only at about 15% of outermost radii where the neoclassical fluxes are strongly decreased due to the low temperature a simple anomalous contributions to the heat fluxes are added. The assumption of mainly neoclassical transport leads to an upper limit for the bootstrap current and temperatures. The modeling performed for 10 MW X2-mode ECR heated plasma of 10^{20} m^{-3} density has shown that the bootstrap current has about 20% lower value in all important magnetic configurations (*low-mirror, standard, and high-mirror*) as compared with the calculation done without momentum conservation. The bootstrap current is maximal in case of the *low-mirror* configuration; it has value I_b=88 kA with momentum correction and I_b =104 kA without. For the *high-mirror* case, the bootstrap current with momentum correction is I_b =19.5 kA and I_b =28.4 kA without. As for the ECRH driven current (ECCD), momentum conservation at the kinetic level was already present in previous calculations. New in the simulation of ECCD was using the weakly relativistic approach in the Spitzer function calculations and the generalized formulation for the trapped-particle fraction f_{tr} meaning that increasing of collisionality leads to a decrease of f_{tr}. The simulations have shown that the value of ECCD is weakly affected by including the generalized formulation for f_{tr}, at least for electron temperatures in the range 1-8 keV. More important are the relativistic effects which lead, however, to a small decrease of ECCD of order ≈1%.

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Benchmark of neoclassical thermal transport matrix

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The international collaboration on neoclassical transport in stellarators [1] has been established for testing and benchmarking various theoretical tools and approaches used to calculate neoclassical transport coefficients. Within the local ansatz and proper ordering scheme it is possible to reduce a 5D drift kinetic equation to a 3D problem and characterize all neoclassical effects in terms of three *mono-energetic* coefficients describing the radial transport, the bootstrap current and the parallel conductivity. These coefficients depend on the flux surface label and two dimensionless parameters: the normalized radial electric field and collisionality. Benchmarking of the mono-energetic transport coefficient has been successfully completed; as an outcome of these efforts *neoclassical databases* have been created for several stellarators using various approaches including Monte Carlo simulations and numerical solutions of the ripple-averaged and drift kinetic equations. However, in order to use the mono-energetic coefficients in the predictive transport codes and for the analysis of experimental results, the full neoclassical (or thermal) transport matrix must be obtained through the database interpolation and the appropriate convolutions of resulting quantities with a local Maxwellian. In this paper, we present benchmarking results of two techniques used for creating the full neoclassical transport matrix. The testing procedure, for example for the LHD heliotron, is as follows. At first we perform some “theoretical” experiment for LHD by applying the predictive transport code [2] with neoclassical transport provided by the convolution based on the database created from DKES [3] simulations of this device and conventional interpolation in 3D parameter space. Then the resulting plasma profiles are analyzed by means of neoclassical database DCOM/NNW [4] which implements the Monte Carlo code for creating discrete data set and neural network technique for interpolation during convolutions with a local Maxwellian. The outcome of transport analysis, the derived thermal diffusion coefficients and the radial electric field, are compared with the original prediction; good agreement of both results is found despite the quite different approaches used. The same method has been applied for the comparison of the predicted transport coefficients for the TJ-II stellarator with ones resulting from transport analysis based on the MOCA [5] (another Monte Carlo code) database and neural network technique for interpolation of the database results.

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Neoclassical and anomalous flows in stellarators

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The impact of magnetic geometry and plasma profiles on flows and viscosities in stellarators is investigated. This work examines both neoclassical and anomalous flows for a number of configurations including a particular focus on the Helically Symmetric Experiment (HSX) and other quasi-symmetric configurations. Neoclassical flows and viscosities are calculated using the PENTA code [1,2]. For anomalous flows, the neoclassical viscosities from PENTA are used in a transport code that includes Reynolds stress flow generation [3]. This is done for the standard quasi-helically symmetric configuration of HSX, a symmetry-breaking mirror configuration and a hill configuration. The impact of these changes in the magnetic geometry on neoclassical viscosities and flows in HSX are discussed. Due to variations in neoclassical viscosities, HSX can have strong neoclassical flows in the core region. In turn, these neoclassical flows can provide a seed for anomalous flow generation. These effects are shown to vary as the ratio of electron to ion temperature varies. In particular, as the ion temperature increases relative to the electron flow shear is shown to increase.

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A convergence study for the Laguerre expansion in the moment equation method for neoclassical transport in general toroidal plasmas

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A dependence of the neoclassical parallel flow calculations on the maximum order of the Laguerre-polynomial expansions [1] is investigated in a magnetic configuration of the Large Helical Device [2] with using the mono-energetic coefficient database obtained by an international collaboration [3]. Based on a previous generalization (of so-called Sugama-Nishimura method) to arbitrary order of the expansion [1], the 13M, 21M, and 29M approximations are compared.
In the previous comparison [1], only the ion distribution function in the banana collisionality regime of single-ion-species plasmas in tokamak configurations is investigated. However, the energy-dependence of the mono-energetic viscosity coefficients in helical/stellarator configurations is not so simple than that in tokamaks, and the present experiments are conducted in wide range of parameters (n_e, T_e, T_i, Z_{eff}).
In this presentation, the dependence in the problems including the electron distribution function in general collisionality regime in an actual non-symmetric toroidal configuration will be reported. Especially, qualities of approximations for the electron distribution function are investigated in detail since it is important in many aspects in stellarator confinement studies, such as the radial electric field [4-5], the MHD equilibrium [6], and its control scenarios [7].

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A Monte-Carlo-based calculation of neoclassical flows and viscosity for nonaxisymmetric toroidal plasmas

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Accurate computation of neoclassical flows and viscosity is an important issue to investigate the momentum transport in toroidal plasmas. We have developed the method to evaluate the required diffusion and viscosity coefficients (L, M, N) using a guiding-center Monte Carlo code[1]. Here, we apply this code to evaluate the plasma flows and viscosity for nonaxisymmetric toroidal plasmas in the framework of moment-equation approach[2].

In the first part of this paper, we outline the method to evaluate the viscosity and diffusion coefficients for arbitrary collisionality by the guiding-center Monte Carlo code. We employ techniques in molecular dynamics simulations, which evaluate the Navier-Stokes transport coefficients[3]. The diagonal and off-diagonal transport coefficients are evaluated from Einstein relations, which generalize the concept of the mean-square displacement for the self-diffusion coefficient into all the elements of transport matrix. We here note that the moment-equation approach[2] allows us to circumvent the difficulty in treating the momentum conservation in the level of kinetics. We show benchmarking results with the DKES code[4] for an $l = 2$ magnetic-field model with and without incompressible radial electric fields. The validity of calculated neoclassical viscosities are limited only by the conventional neoclassical ordering.

In the second part, we present the calculation of the neoclassical plasma flows for given plasma profiles. We combine the above numerical code with the moment-equation approach. The viscosity and diffusion coefficients L, M, and N specify the viscosity-flow relations; the neoclassical plasma flows are then evaluated from the algebraic solutions of moment equations. The essential feature of our code is that it is capable of calculating all the elements of the diffusion and viscosity coefficients. These coefficients are necessary to obtain the toroidal and poloidal plasma flows from the numerically evaluated poloidal and toroidal viscosities. Similar treatments have only been performed with the DKES code so far[5].

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Development of neoclassical transport module for the integrated simulation code in helical plasmas: TASK3D

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In helical systems, the neoclassical transport is one of the important issues in addition to the anomalous transport, because of a strong temperature dependency of heat conductivity and an important role in the radial electric field determination. Thus the development of a reliable tool for the neoclassical transport analysis is necessary for the transport analysis in helical plasmas. We have developed a neoclassical transport database for LHD plasmas, DCOM/NNW[1]. The mono-energetic diffusion coefficients are evaluated based on the Monte Carlo method by DCOM code and the mono-energetic diffusion coefficient database is constructed using a neural network technique. The input parameters for the database are the collision frequency, the radial electric field, the minor radius and the configuration parameters (R_{axis}, β_0). Recent increment of heating power raises the plasma temperature in LHD. Because the collision frequency decreases in proportion to $T^{3/2}$, we have to estimate the diffusion coefficient in the more collisionless regime. However, previous DCOM code requires huge calculation time to obtain the diffusion coefficient in such collisionless regime.

In this paper, we improve the DCOM code to reduce the computation time in order to obtain the mono-energetic diffusion coefficients in the more collisionless regime. As a result the DCOM calculation becomes about 6 times faster than that of the previous version. Also we apply GSRAKE[2] code, which solves the ripple-averaged drift kinetic equation, to obtain further collisionless regime. Finally we construct a neoclassical transport database DCOM-GSRAKE/NNW for LHD (DGN/LHD). The neoclassical transport analyses of high temperature LHD plasma are done using DGN/LHD. And we newly apply DGN/LHD as a neoclassical transport analysis module to TASK/3D[3], which is the integrated simulation code in helical plasmas, and study the role of the neoclassical transport in several typical LHD plasmas.

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Comprehensive Understandings on the Effects of Radial Electric Field on the Reduction of Neoclassical Diffusion in Improved Confinement in LHD

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Several improvement confinement have been achieved in the LHD experiment, such as , high ion-temperature (Ti) plasmas [1], high electron-temperature (so called core electron-root confinement, CERC) plasmas [2], and the high density plasmas with the establishment of the internal diffusion barrier (IDB) [3]. In the mean time, it is of great importance to suppress the neoclassical (NC) diffusion since the NC particle and heat diffusions are expected to increase as the collisionality is decreased in helical devices such as LHD without the presence of the radial electric field (ripple diffusion). It has been experimentally demonstrated that the radial electric field (Er) can be well interpreted by the NC ambipolarity [4] (that is, $\Gamma e=\Gamma i$, where $\Gamma e,i$ denotes the electron and ion NC particle flux, respectively). The above-mentioned extensions of parameter regime in density and temperature in LHD have provided further opportunities to validate such interpretation, and to make the extrapolation towards reactor-relevant parameter regime more reliable.

In this regard, the comprehensive understandings have been tried to be elucidated from the ambipolar Er analysis in a wide range of density and temperature regimes, based on the NC transport calculations by using the GSRAKE code [5]. The high-Ti, high-Te and IDB/SDC (super dense core) plasmas are characterized by the different temperature ratio and density regime (high-Ti: Ti/Te>1, on the order of 10^{19} m^{-3}, high-Te: Te/Ti>>1, on the order of 10^{18} m^{-3}, and IDB/SDC: Te (should be ~ Ti), on the order of 10^{20-21} m^{-3}), which will be comprehensively compared with the predictions of NC ambipolar Er in a wide range of parameters.

Based on such comparisons in different parameter regimes, the appropriate scenarios for further extension of plasma parameters (individually, and even simultaneously) in the near-term LHD experiments, and, moreover, in reactor-relevant heliotron plasmas by keeping at least NC diffusion at small level, can be obtained.

These issues are main objects of the paper, and will be reported in more detail in the workshop.

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Influence of low order rational surfaces on the radial electric field of TJ-II ECH plasmas

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During the 2008 experimental campaign a new mode of operation was implemented on TJ-II Heliac [1]. In the new mode of operation currents in the coils can be ramped during the discharge in order to change the magnetic field configuration. This allows for dynamic scan of $\iota\!\!\!-$ in a wide range during a single shot and can reveal effects that are difficult to notice in scans performed on a shot to shot basis [2].

In January 2009 the fast frequency hopping reflectometer installed in TJ-II was upgraded to a Doppler reflectometer [3]. The new system allows for measurements of the perpendicular rotation velocity of density fluctuations and their fluctuation level with excellent temporal and spatial resolution.

The new mode of operation together with the Doppler reflectometer system is a unique tool to investigate the dependence of plasma rotation, radial electric field and turbulence on the rotational transform.

Dynamic configuration scans experiments were conducted in order to study the influence of low order rational surfaces on the radial electric field of low magnetic shear ECRH plasmas. It was discovered that the main magnetic resonances (7/4 and 5/3 in this work) always make a positive contribution to the local radial electric field, and the contribution depends on the plasma density and the plasma radius. A local reduction in the level of density fluctuations due to the influence of low order rational surfaces was also observed, and the degree of reduction depends on plasma density. The reduction in the fluctuation level might be due to the radial gradients in the $E \times B$ flow. Coherent modes of two types (low and high frequencies) in the radial electric field due to the rational ***5/3*** were revealed. High frequency modes are also present in the signals from other diagnostics (Mirnov coils, Langmuir probes) and might be related to some MHD activity induced by the rational surface.

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CALCULATION OF NEOCLASSICAL TOROIDAL VISCOSITY IN TOROIDAL PLASMAS WITH BROKEN TOROIDAL SYMMETRY

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A new numerical scheme to evaluate neoclassical toroidal viscosity (NTV) in non-axisymmetric magnetic field configurations like heliotron or tokamak with small perturbation field is presented. NTV has recently been paid much attention, since it relates to topical issues such as stability of resistive-wall-modes and controlling edge-localized-modes by perturbation field coils [1, 2]. We have developed the numerical code FORTEC-3D to solve neoclassical fluxes and simulate the formation of ambipolar electric field in helical configurations[3]. It uses the δf Monte-Carlo method [4] and can treat the effect of the detailed structure of magnetic field to the guiding-center motion and transport, including the finite-orbit-width effect. Since the pressure anisotropy $\delta P \equiv P_{\perp} + P_{\|}$, which is required to evaluate NTV, is obtained from the perturbed distribution function δf , FORTEC-3D can be directly applied to NTV calculation. However, the perturbation of the magnetic field considered here is very weak, $\delta B / B_0$ ~10^{-3}-10^{-4}, than the helical ripple amplitude in helical devices which FORTEC-3D has been applied to. In order to reduce the numerical noise, we adopt the numerical method developed in Ref. [5], in which δP is expanded in Fourier series on a flux surface. In this presentation, NTV calculation is benchmarked in a helical plasma which models LHD plasma, of which the NTV can be estimated from well-established neoclassical theory, as the first step of the application. We will also show test results in a simple tokamak configuration with small perturbation field and analyze the sensitivity of the NTV to perturbation amplitude and how much it is affected by the radial electric field. Broken axisymmetry means that the neoclassical transport is not intrinsic ambipolar and the radial electric field is evolved so that the ambipolar condition $\Gamma_i = \Gamma_e$ is satisfied as it happens in a helical plasma. However, the neoclassical ion flux induced by the non-axisymmetry is proportional to the ion NTV and is known to be sensitive to the radial electric field strength in helical cases. Taking advantage of FORTEC-3D code, the effect of the radial electric field on the NTV in weakly-perturbed tokamak will also be investigated.

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Spectrally resolved Motional Stark-Effect Polarimetry at ASDEX Upgrade

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Affecting magnetic equilibria, the diamagnetic effect is also reflected in motional Stark effect spectra. Giving thereby experimental access to magnetic equilibria in principle, the application on stellarators requires a thorough assessment of the measurement capabilities. The conventional MSE diagnostic measures the polarisation properties of one Doppler shifted Balmer alpha emission line. In contrast the presented diagnostic setup uses the full spectral information in particular the line splitting. This spectrum is dominated by the Stark effect due to the $\vec{v} \times \vec{B}$ Lorentz electric field experienced by the fast beam atoms and hence allows the detection of Stark-multiplets and determination of the local magnetic field. The setup and results will be presented and next improvements discussed. The investigations ASDEX Upgrade serve to assess a possible motional Stark effect diagnostic for Wendelstein 7-X.

Ion internal transport barrier in Large Helical Device

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In Large Helical Device (LHD), the ion internal transport barrier (ITB) appears when the P-NB is injected before the N-NB but not when the N-NB is injected before the P-NB, even if the power of the P-NBI and the N-NBI are identical later in the discharge. The T_e/T_i ratio at the time both P-NB and N-NB are injected is larger in the discharges with prior N-NBI rather than the discharges with prior P-NBI. These observations suggest that the high T_e/T_i ratio contributes to prevent the formation of the ion ITB in LHD. Therefore it is important to keep the T_e/T_i ratio close to or below unity at the onset of the high power NBI to achieve a high ion temperature. There are two approaches to achieve high ion temperature plasmas by keeping the T_e/T_i ratio at a low level in LHD. One is to perform the P-NBI injection only before the start of high power heating, which gives the target plasma with a low T_e/T_i ratio. Another approach is the injection of hydrogen or carbon pellet before the start of high power heating, which results in a T_e/T_i ratio close to unity. The carbon pellet has an additional benefit in increasing the power deposition of NBI to ions through ion-impurity collisions. At the formation of ion ITB, the ion temperature gradient starts to increase at approximately half of the plasma minor radius (up to ~ 9 keV/m) then the ion temperature gradient farther inside increases later.

As seen in Fig.1, the ion thermal diffusivity gradually decreases towards the plasma center in the L-mode phase (t = 2.10s). After the formation of the ITB, the reduction of the thermal diffusivity is observed in a wide interior region of the plasma ($0.25 < \rho < 0.75$). The drop of the thermal diffusivity has a maximum at a half of the plasma minor radius and the reduction of the thermal diffusivity is by a factor 2 at t = 2.23s (just after the formation of ITB). In this phase, the thermal diffusivity even slightly increases outside the ITB region ($\rho < 0.25$ and $\rho > 0.75$). Later in the steady-state phase of ion ITB, the ITB region expands towards the plasma center and the reduction of the thermal diffusivity is observed near the plasma center ($\rho \sim 0.2$). In the ion ITB in LHD, the large ion temperature gradient region expands towards plasma center in time and becomes wide in the steady-state phase, which is in contrast to the ITBs in tokamaks, where the large ion temperature gradient region moves towards the plasma periphery in time and becomes narrow in the steady state phase.

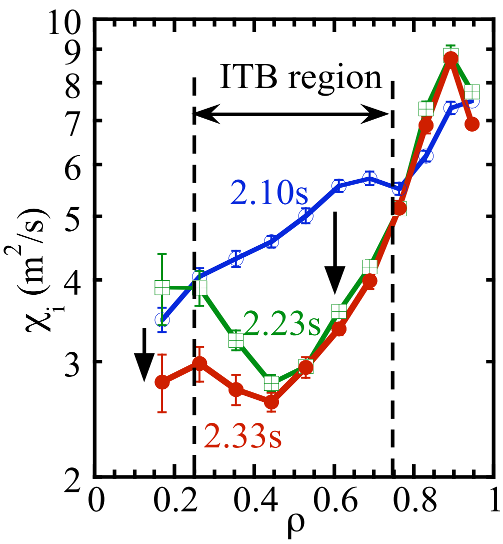

Fig.1 Time evolution of radial profiles of ion thermal diffusivity at t = 2.10s (in the L-mode phase), t = 2.23s (just after the formation of ITB) and t = 2.33s (in the steady-state phase of ITB).

Considerations from the viewpoints of neoclassical transport towards higher ion-temperature heliotron plasmas

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Experiments have been conducted in LHD to extend ion-temperature (T_i) regime, to achieve 5.6 keV in the experimental campaign of FY2008 [1]. In such high-T_i plasmas, the gradient of T_i is increased, and typically T_i is higher than the electron temperature (T_e). The negative radial electric field (E_r) (ion root) has been predicted there, based on the neoclassical ambipolarity, and it has been experimentally verified with the potential measurement by utilizing the heavy ion beam probe (HIBP) [2] even in a circumstance of the formation of the impurity hole [3].

The E_r can reduce the neoclassical (NC) transport in low-collisional LHD plasmas as estimated in Ref. [4]. However, it is increasing as T_i is increased even with the presence of E_r, although the degree of the increase is much smaller than that predicted from the "pure (E_r=0)" $1/\nu$ ripple transport [5]. Thus, it is important to consider plausible approaches to reduce (at least) the NC ion diffusion for the further extension of T_i-parameters towards reactor-relevant situations.

One of the plausible approaches is to enhance the ion-root E_r, which can be done by increasing T_i/T_e ratio [6], which experimentally corresponds to the situations of the predominant ion heating. This approach would be confirmed with the increase of ion-heating power by the additional NBI (5th beam line).

Another approach, with which much more reduction can be anticipated, is the realization of the electron-root E_r. The electron-root E_r with larger absolute values of E_r compared to that of the ion-root E_r can further reduce NC diffusion, as well known. Artificial parameter-scan NC calculations have revealed that when T_e is also increased (even with keeping T_i/T_e ratio), at certain value of T_e, the electron-root E_r can appear to further reduce NC ion diffusivity compared to that expected to the ion-root E_r. It was also demonstrated that, for reactor-relevant parameters (T_i~T_e~10 keV [T_i/T_e=1], density of about $10^{20}m^{-3}$), the electron-root E_r can be possible due to the enhancement of ripple transport both for the ion and the electron (nonlinearity of the dependence of the NC fluxes on E_r is increased). In this approach, T_e should be also increased requiring the heating power deposited to the electrons.

These two approaches will be systematically investigated to elucidate the plausible ways to further increase, especially T_i parameters, in LHD, and in (a scaled-up model) heliotron reactor as well.

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ELM triggering by non-axisymmetric magnetic fields in NSTX

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The application of non-axisymmetric magnetic fields has been shown to destabilize edge-localized modes (ELMs) in the National Spherical Torus Experiment. The perturbation is applied using a set of midplane coils external to the vacuum vessel, configured to produce a dominantly n=3 field. Calculations assuming vacuum perturbation fields show that a strongly stochastic edge layer is formed by the applied field, with a width of $\Delta\psi_N \sim 0.4$. Simulations of the plasma response using the Ideal Perturbed Equilibrium Code (IPEC), however, indicate a strong attenuation of resonant components of the perturbation in the plasma core, so that the width of the equivalent stochastic layer is reduced to $\Delta\psi_N \sim 0.1$. When applied during a period of a discharge that exhibits very small ELMS or is entirely ELM-free, the perturbation causes large ELMs to begin within 50 ms. The stongest change in the core plasma profiles is a global decrease in the toroidal rotation due to magnetic braking. Measurements of the pedestal kinetic profiles taken between the application of the n=3 field and the ELM onset show little change in the density, an increase in the pedestal electron temperature, and a corresponding increase in the peak pressure gradient. MHD stability has been calculated using the PEST code, and show that the plasma is stable before the n=3 field is applied, and unstable to an edge mode after the n=3 field is applied. The triggering effect of the 3D fields has been used to controllably introduce ELMs into high confinement ELM-free H-modes enabled by lithium wall conditioning, which typically show strong impurity accumulation. Using 3D fields to perform ELM pacing during these discharges led to a reduction in impurity accumulation, while maintaining high energy confinement. Large amplitude perturbations allow rapid triggering, so that the frequency of the ELMs can be increased to reduce their size. While the ELM size decreases at higher frequencies, low frequency triggering may be optimal for controlling impurity buildup with minimal impact on energy confinement. This technique is being optimized, so that the combination of lithium wall coatings to improve energy confinement and ELM pacing with applied non-axisymmetric fields can be developed to produce stationary, high-performance plasmas.

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Ideal Perturbed Equilibria in Tokamaks and Control of External Magnetic Perturbations

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Tokamaks are sensitive to deviations from axisymmetry as small as dB/B ~ 0.01% with significant degradation or improvement in performance. Ideal Perturbed Equilibrium Code (IPEC) [1] has been developed in order to understand plasma responses to such a small nonaxisymmetric perturbation. IPEC has resolved paradoxical error field problems in present tokamaks [2,3], and is being used to characterize Resonant Magnetic Perturbations for the suppression of Edge Localized Modes.

The calculations of perturbed equilibria, such as IPEC in tokamaks, can be used to decompose the distributions of external magnetic perturbations on a control surface by their importance to critical plasma properties. The coupling between the external field at the control surface and the resonant field driving islands at the rational surfaces determines the most important distribution of external magnetic field on the control surface for island opening and plasma locking. It has been shown that the most important distribution of external magnetic field changes little across various plasma profiles and parameters [4]. The other distributions of external magnetic field on the control surface are less important by an order of magnitude. The coils required to control certain physics properties, such as the formation of magnetic islands or the breaking of quasi-symmetry, can be relatively easily designed in tokamaks and stellarators if a perturbed equilibrium analysis is carried out to investigate the important distributions of external magnetic field that determine the changes in that property. This presentation will give (1) brief introduction to IPEC and its applications, (2) the dominant and optimized external field on a control surface based on the coupling with the resonant field driving islands in tokamaks, and (3) implications to the design of control coils in tokamaks and stellarators.

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Self-organized helical equilibria in the RFX-mod reversed field pinch

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With operation at high plasma current (I ≈ 1.5 MA), plasmas in the RFX-mod reversed field pinch reproducibly self-organizes in a 3d, non-axisymmetric single helical axis equilibrium with helicity m=1, n=7 (which corresponds to the innermost m=1 rational surface for the RFX-mod safety factor profile) [1]. This helical state has almost conserved magnetic flux surfaces, which have been interpreted as ghost surfaces [2] leading to strong core electron transport barriers. Electron temperature Te reaches 1.3keV @1.7MA. Ti is ~(0.5-0.7)Te, consistently with ion collisional heating. The core barrier extends up to ~0.65r/a. Magnetic surfaces quality improves with plasma current, thanks to the simultaneous decrease of magnetic chaos and increase of the helical field strength. The persistence of these states also increases with plasma current.

An equilibrium reconstruction based on a Newcomb equation solver in toroidal geometry allows for mapping of the electron temperature, density and SXR emissivity profiles on helical flux surfaces thus proving the correlation between kinetic plasma quantities and the underlying helical magnetic topology. Indeed the helical flux surfaces are found to be isobaric. The magnetic topology of a SHAx state has been also reconstructed by the 3d VMEC equilibrium code, which is being adapted for the RFP.

The location of electron transport barriers is correlated with the safety factor *q* profile of the helical equilibrium, similarly to what is observed in Stellarator and Tokamak. A region of zero magnetic shear is a key ingredient to trigger internal transport barriers.

Main gas particle confinement time improves in pellet-fuelled plasmas, with record value ~10ms. No evidence of core impurity accumulation comes from Laser Blow Off experiments.

The injection of pellets (with different penetration lengths) into SHAx states has been used to study the effect of the helical topology on particle transport showing also that no fast poloidal homogenization is taking place in these helical states. First numerical studies on neoclassical transport in helical RFP states have been performed by means of the ORBIT code.

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Radial and Parallel Transport Fast Camera Observation on LHD SOL Region

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Large Helical Device (LHD) has steadily progressed during recent years in the achievement of greater β plasmas, of great importance for the realization of an efficient fusion reactor, reaching $<\beta_{dia}>$ values up to 5% in recent experiments. During this process, a great deal of effort has been dedicated to several tasks, such as understanding their MHD behaviour [1]. However, LHD plasmas at these high β conditions represent also an optimal set up for the observation of edge and SOL transport mechanics with ultra fast visible cameras: thanks to the geometry and size of the chamber, and the well defined divertor leg surfaces, non-integrated observation of the SOL channel is possible at sampling times (20 μs) still relevant for transport time scales.

During 2008 NIFS experimental campaign, a new fast camera system was installed on the LHD stellarator, following previous work performed on LHD with conventional CCD cameras [2],[3] and previous work performed on TJ-II with similar camera systems [4]. Among several other experiments, the observation of edge transport mechanics during high β (up to 5%) discharges (R_{ax}=3.6 m, B_t=-0.425 T, B_q=100%, γ=1.197) was undertaken. By this observation, filament-like structures were found propagating along the SOL in radial and parallel directions. These high β plasmas are obtained by MHD optimization of the magnetic configuration [1] and their edge is characterized by the presence of the $\iota/2\pi = 3/2$ surface almost outside the LCFS (r_{eff}=0.99) [5]. The analysis and comparison of camera and magnetic probes arrays signals shows the mode-related origin of some of the filaments and suggests a strong link between SOL and divertor transport, and edge MHD activity. Besides, filaments originate in the edge region of the plasma and travel along X points trails in the parallel direction and radially towards the wall, where sometimes seem to be reflected. This reflection effect is explained in terms of plasma wall interaction mechanics (with good agreement with previous experiments [6]), providing a direct link between the plasma release at the edge and the flux to the plasma facing components.

As well, by means of several image analysis techniques [4],[7], both parallel and radial velocity components are measured (again, with good agreement with previous literature [8]), leading to a discussion about transport time scales in each direction. Finally, the presence of resistive pressure gradient driven turbulence (characteristic of high β operation in LHD) suggests the possibility of Self Organized Criticality behaviour on the edge. Therefore, the ejection pattern of the filament structures is analyzed by specific statistical methods and the results are discussed.

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Peripheral turbulence measurement in Heliotron J using fast cameras

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Peripheral turbulence should be related to plasma confinement, and to understand it is very important issue to realize the nuclear fusion reactor effectively. In particular, the control of turbulence during additional heating should be required in near future even in the small experimental device such as Heliotron J.

In this study we report the peripheral turbulence measurement using Langmuir probes, magnetic probes, and fast cameras[1-4]. Topics are turbulence behavior during ECH(70GHz, 450kW), NBI(30keV,700kW), and ICRF(19-22MHz,200kW). ECH is used to get the first discharge in Heliotron J plasma. In general turbulence was easily observed during additional heating (or during high power heating). However, it was found that turbulence was suppressed under some conditions during ICRF and/or NBI. Also, the super molecular beam injection (SMBI)[5-7] is used in Heliotron J plasmas to increase the electron density and to reduce the total recycling. In SMBI experiment even the electron density raise over the threshold density to the H-mode, the transition does not occur frequently. Peripheral turbulence behavior seems to be “dithering to transit from the L-mode to the H-mode” before and after SMBI. In helical system the radiation collapse occurs sometimes in the H-mode, and it is not favorable phenomenon. Therefore, it is very important to understand the relationship between peripheral turbulence behavior and the H-mode.

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Overdense Plasma Operation in WEGA Stellarator

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The application of electron cyclotron resonant heating (ECRH) for high density and high beta operation in magnetically confined plasmas is limited due to reflections of the heating wave at associated cut-off limits that prohibits a propagation of the heating wave into the central plasma region. However, overdense operation in stellarators [1] and high beta regime in spherical tokamaks, can be achieved by an alternative heating concept that bases on the conversion of an incident electromagnetic wave into an electrostatic Bernstein wave (EBW). At WEGA stellarator experiments in overdense argon and helium plasmas fully sustained by electrostatic Bernstein waves have been performed. WEGA is a classical five period $l = 2$ stellarator with a very flexible magnetic configuration allowing operation at 0.5 T for about 20 s.
For the generation of EBWs a 28 GHz ECRH system (10 kW cw) and an associated transmission and mirror system were applied allowing a two-step conversion process at the plasma edge from second harmonic electromagnetic O- to X-waves and a subsequent conversion at the upper hybrid layer into electrostatic Bernstein waves which may propagate into the plasma without a density limit. The conversion efficiency does strongly depend on the angle between the incident wave and the magnetic field vector and should be optimum for an oblique angle of 55°. However, a moveable mirror-system was installed that allows a variation of this angle.
EBWs can only propagate inside the plasma above a density threshold (O-mode cut-off) of $n_e = 1 \times 10^{19}\,\mathrm{m}^{-3}$. To reach this density additional heating using the Ohmic transformer or non-resonant 2.45 GHz magnetron ECH (26 kW cw) was necessary. However, in recent experiments with argon plasmas the density threshold could also be obtained by means of the 28 GHz ECRH system only. The density can be determined by a single-channel 80 GHz interferometer measuring the line averaged density and a Langmuir-probe installed on a fast reciprocating manipulator.
While the typical radiation and electron temperature during second harmonic O or X-mode heating are in the order of a few 10 eV only, a radiation temperature of up to more than 10 keV could be observed in the central region of the plasma by a 12-channel radiometer when reaching the OXB state [2]. This result could be confirmed by soft x-ray measurements obtained with a pulse height analyzer. On a 12-channel bolometer also a strong increase of the radiated power on the central channels was visible during the OXB state. Furthermore, a sniffer probe, which measures the fraction of the non-absorbed 28 GHz stray radiation, showed a significant drop when reaching the threshold density indicating an improved absorption of the incident waves. It is assumed, that the observed central temperature of 10keV is due to a small fraction of super thermal electrons in the central region heated by the EBWs, while the bulk electrons still have temperatures of a few 10 eV only.

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Overview of results from the CNT stellarator and future plans

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An overview of recent numerical and experimental results from the Columbia Non-neutral Torus (CNT) will be given. CNT is a simple, ultralow aspect ratio stellarator dedicated to studies of non-neutral and electron-positron plasmas. The non-neutrality and low density of the plasmas in CNT allows one to study neoclassical transport in the strong electric field regime [1]. CNT operates with a surplus of electrons – most of the time with only a trace amount of ions (n_i/n_e <1%), creating negative space potentials of several hundred volts despite having T_e on the order of 5 eV [2]. This allows the study of confinement with negative radial electric fields much larger than those that would arise from ambipolarity constraints. CNT is a classical stellarator with a large fraction of magnetically trapped particles that would ordinarily leave in about 10^{-5} seconds if it weren't for the ExB drift. Due to the strong electric field, the ExB drift dominates over the magnetic drifts in CNT. Instead, numerical simulations show that deeply trapped particles are well confined. However, unconfined particle orbits exist in CNT for reasons largely unrelated to magnetic drifts. These include bad orbits due to toroidal resonances, and bad orbits due to the mismatch between electrostatic potential surfaces and magnetic surfaces. Although modest parallel electric fields are intrinsic to the pure electron equilibria in CNT [3, 4], much larger variations of electric potentials can be created on outer surfaces by external charged objects. A segmented conducting mesh was installed in 2007 to control the electrostatic boundary conditions in CNT. The mesh improved confinement considerably, and a new mesh is being designed now [5]. Studies of plasmas with a non-negligible ion fraction show the appearance of a coherent oscillation that appears to break parallel force balance of the electron fluid [6]. Increasing the ion content further, a systematic study of behavior as the degree of neutralization is varied from pure electron to quasineutral has begun [7].
The current plans for the electron-positron plasma experimental phase will be presented.

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Wednesday

Oct. 14, 2009

08:30 – 09:15	PL02	K.H. Finken	***Role of Resonant Magnetic Perturbations in Tokamaks***
09:15 – 09:45	I16	M. Jakubowski	***Influence of the magnetic topology on transport in the plasma boundary with resonant …***
09:45 – 10:15	I17	E.A. Unterberg	***Particle Exhaust and Scrape-off Layer Conditions During Resonant Magnetic Perturb. …***
10:15 – 10:45	I18	W.M. Solomon	*Effect of Non-axisymmetric Fields on Toroidal Rotation Dynamics*

Coffee

11:15 – 11:45	I19	F.L. Tabares	*Energy and particle balance studies in the full lithiated TJ-II stellarator*
11:45 – 12:15	I20	S. Masuzaki	*Edge heat transport in the helical divertor configuration in LHD*
12:15 – 12:45	I21	W.X. Ding	*Particle Transport due to Stochastic Magnetic Field in a High-Temperature Plasma*

Lunch

13:45 – 15:40	P3-01	Y. Feng	***On Particle Control of the W7-X Divertor***
13:45 – 15:40	P3-02	B. Nold	*Inter-machine edge turbulence data base*
13:45 – 15:40	P3-03	Canik	*Divertors for quasi-symmetric stellarators*
13:45 – 15:40	P3-04	J. Geiger	***Effects of Toroidal Currents on the Magnetic Configuration of W7-X***
13:45 – 15:40	P3-05	J. Geiger	***Magnetic Diagnostics and Plans for Equilibrium Reconstruction at W7-X***
13:45 – 15:40	P3-06	M. Gobbin	***Numerical reconstruction of spontaneous helical equilibria in RFX-mod***
13:45 – 15:40	P3-07	S. Sakakibara	***Remarks on Finite Beta Effects in International Stellarator/Heliotron Scaling***
13:45 – 15:40	P3-08	S. Sakakibara	***Study of MHD Characteristics by Magnetic Axis Control in high-beta plasmas of LHD***
13:45 – 15:40	P3-09	F. Watanabe	*Characteristics of edge MHD modes and ELM activities observed in LHD plasmas*
13:45 – 15:40	P3-10	S. Okamura	***Modification of LHD magnetic configurations based on the boundary shape control for …***
13:45 – 15:40	P3-11	M. Sato	***Simulation study of MHD stability beta limit in LHD by TASK3D***
13:45 – 15:40	P3-12	M. Schlutt	***Investigation of equilibrium plasma beta limits in 3-D magnetic topologies***
13:45 – 15:40	P3-13	N. Yanagi	***Design integration on split and segmented-type helical coils for FFHR***
13:45 – 15:40	P3-14	T. Brown	***Improved W7-X current lead temporary support and assembly arrangement***
13:45 – 15:40	P3-15	T. Dodson	***Configuration space control for Wendelstein 7-X***
13:45 – 15:40	P3-16	C.B. Deng	*Energetic-Electron-Driven Instability in the Helically Symmetric Experiment*
13:45 – 15:40	P3-17	E.D. Fredrickson	***Modeling Fast Ion Transport in TAE Avalanches in NSTX***
13:45 – 15:40	P3-18	H. Okada	*Heating Position Dependence of Energy Spectra of Fast Ions Generated by ICRF …*
13:45 – 15:40	P3-19	S. Kubo	***Collective Thomson Scattering of 77 GHz High Power ECRH Beam in LHD***
13:45 – 15:40	P3-20	R. Seki	***Monte-Carlo Study of Perpendicularly Injected High-Energy Particles using Real…***
13:45 – 15:40	P3-21	N. Pomphrey	***Improving the tokamak and RFP concepts by addition of 3D fields (and use of stellarator …***
13.45 – 15:40	P3-22	W.F: Bergerson	*Non-Axisymmetric Helical Structures in the MST Reversed-Field Pinch*
13:45 – 15:40	P3-23	T.E. Evans	***Overview of the DIII-D non-axisymmetric center post coil design and physics basis***
13.45 – 15:40	P3-24	S. Satake	*Calculation of neoclassical toroidal viscosity in toroidal plasmas with broken …*

15:40 – 16:10	I22	S. Masamune	*Transition to helical RFP state and associated change in magnetic stochasticity in …*
16:10 – 16:30	C12	P.J. Heitzenroeder	*NCSX construction accomplishments and lessons learned*
16:30 – 16:50	C13	T. Gotot	*Core Plasma Design of a Heliotron Reactor*
16:50 – 17:10	C14	F. Schauer	*Extrapolation of the W7-X magnet system to reactor size*
17:10 – 17:30	C15	P.R. Garabedian	*QAS design of the DEMO reactor*
17:30 – 17:50	C16	G.H. Neilson	*Stellarator configuration improvement using high temperature superconducting monoliths*

Role of Resonant Magnetic Perturbations in Tokamaks

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Resonant magnetic perturbations (RMPs) are imposed in tokamaks by external coil currents. Perturbations near the plasma edge can cause stochastization of the edge magnetic field and open magnetic field lines to the wall. Fundamental aspects of stochastization have been studied since the early 1980’s in several tokamaks such as TEXTOR [early modular version and later the Dynamic Ergodic Divertor (DED)], TEXT, Tore Supra, JFT-2M, and Hyptok. Stochastization of the plasma edge is intended to provide a layer of enhanced transport in order to more easily remove particles and to decouple this layer from the high confinement core.

In particular the opening of field lines to the wall results in a modification of the edge density and temperature of the plasma and to an alteration of the electric field as well as a change of the edge flow pattern, improvements or degradations of the plasma confinement and finally to the highly desired suppression of edge localized modes (ELMs) in high confinement regimes. RMPs are intrinsically 3D in nature and are possibly related to similar effects on confinement in stellarators.

The perturbation field can also interact resonantly with internal magnetic structures and excite meta-stable tearing modes to well developed states which deteriorate the confinement. These modes rotate naturally in the plasma — at least before they are locked to the walls — and therefore a rotating perturbation field is an ideal tool to study the excitation mechanism of those instabilities.

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Influence of the magnetic topology on transport in the plasma boundary with resonant magnetic perturbation at DIII-D

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Stochastic boundaries in fusion devices have been investigated in tokamaks, stellarators and reversed field pinch experiments for many years. However, since edge localized modes (ELMs) have been successfully eliminated in H-mode plasmas at the DIII-D tokamak [1] with small, edge resonant magnetic perturbations, it became a widely investigated topic in tokamaks. At DIII-D stochastic boundaries are produced by coils external to the plasma, which has allowed ELM mitigation over a wide range of pedestal collisionalities and plasma shapes. Experiments on Tore Supra [2] or TEXTOR [3] have shown that a stochastic boundary creates complicated, 3D topologies, with a strong influence on plasma transport. Clear evidence for the stochastization of the plasma boundary has been seen in DIII-D [4,5]. Plasmas with an ITER-similar shape show evidence for the 3D topology of the boundary. Magnetic field there consists of field lines with very different connection lengths. The most obvious manifestation of the perturbed plasma edge is the strike line splitting observable in heat and particle fluxes. At low collisionalities, divertor footprint patterns are in fair agreement with vacuum topology, however at high collisionalities splitting of the separatrix is much stronger, which suggests a plasma amplification process. In H-mode plasma edge radial pressure profiles are a strong function of q_{95} [6], in line with TEXTOR findings [3,6]. Here, an overview of 3D magnetic field structures and transport in H-mode and L-mode plasmas with n=1 and n=3 resonant magnetic perturbations at DIII-D is presented. Many features observed in the DIII-D experiments are generic for all the fusion devices with a stochastic plasma boundary.

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Particle Exhaust and Scrape-off Layer Conditions During Resonant Magnetic Perturbations in Deuterium and Helium Discharges on DIII-D*

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The complete suppression of edge localized modes (ELMs) in a tokamak using the resonant component of a 3D magnetic perturbing field (RMP) has been demonstrated on DIII-D at ITER-similar pedestal electron collisionalities and cross-sectional shapes. Accompanying the suppression is an increase in particle transport leading to the reduction in density across the entire minor radius. It is observed that the pedestal density reduction varies from 2%–30% of the ELMy H-mode density in most discharges but shows no obvious correlation with the magnitude of perturbing coil current. Recent analysis using global particle balance and measurements of the D_α poloidal distribution show that the wall inventory can be strongly affected by changing the average triangularity ($\langle\delta\rangle$) of the plasma (primarily due to changes in δ_{low}). In particular, the analysis shows that at $\langle\delta\rangle \sim 0.3$ the integrated plasma efflux during the RMP is greater than the total number of particles removed by the cryopump system, indicating active wall pumping. Conversely, at $\langle\delta\rangle \sim 0.5$ in a scaled ITER-like shape, the plasma efflux during the RMP is balanced by the cryopump exhaust, i.e., no wall pumping is inferred. Further investigations using vacuum field-line tracing indicate a better alignment of the perturbed, non-axisymmetic separatrix to the cyropump aperture at $\langle\delta\rangle \sim 0.5$ compared to $\langle\delta\rangle \sim 0.3$, which also leads to a bifurcation in edge plasma conditions and divertor pumping between the two shapes. Additionally, the D_α intensity in the $\langle\delta\rangle \sim 0.5$ discharges increases by ~50–100% when compared to $\langle\delta\rangle =0.3$ discharges implying an increase in the scrape-off layer neutral density with $\langle\delta\rangle \sim 0.5$. This observation coupled with an increase in particle exhaust that enhances the capture of efflux particles during the RMP results in an improvement in the density control in ITER similar shape discharges on DIII-D. Comparisons with helium discharges will also be discussed. With helium as the main ion species, the particle wall retention time in graphite is expected to be reduced by many orders of magnitude. Therefore the wall sink/source can be effectively neglected, thus allowing a direct test of the importance of a wall sink in RMP pumpout and ELM suppression physics. Specifically, these results work toward a goal of better understanding the role of particle sources and sinks during the RPM density pump-out. Furthermore, this work demonstrates particle exhaust control and ELM suppression without significant wall pumping, a feature that is desirable in long-pulse reactors with saturated walls.

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Effect of Non-axisymmetric Fields on Toroidal Rotation Dynamics*

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The role of toroidal rotation on the performance of tokamak plasmas has been well-documented, offering improvements to both confinement and stability. Consequently, optimizing rotation by exploiting knowledge of the generation and damping mechanisms of toroidal angular momentum will benefit future burning plasma experiments. Although non-axisymmetric fields have primarily been considered a sink of angular momentum, recent results from DIII-D have shown that in some cases, they may also act as a source. In particular, the application of non-resonant magnetic fields (NRMFs) is shown to drag the rotation to a non-zero “offset” rotation, which is found to be in the direction counter to the plasma current. As such, plasmas with low initial rotation can actually be accelerated by NRMFs. Modeling of the rotation profile following the application of a pre-programmed waveform of n=3 NRMF has been performed for plasmas with both high and low initial rotation. We determine the initial torque profile exerted by the NRMF from the instantaneous change in the angular momentum profile at the time the field is switched on. This torque is the scaled in time by considering the time evolution of the various plasma dependencies of the NRMF torque. The modeled rotation profile evolution obtained by considering angular momentum balance including the NRMF torque shows excellent agreement with the measured profile dynamics. At low rotation, the role of the so-called “intrinsic rotation”, observed in plasmas without any auxiliary momentum input, becomes important. Specifically, we find that it is necessary to include an effective torque associated with the intrinsic rotation in order to obtain satisfactory agreement with the rotation profile evolution. The inferred effective torque profile from such analysis shows good agreement, both in profile-shape and magnitude, with previous measurements obtained by determining the amount of counter NBI torque required to cancel the intrinsic rotation. A key result is that the application of NRMFs does not appear to eliminate or alter the drive for intrinsic rotation.

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Energy and Particle balance studies in the full lithiated TJ-II stellarator.

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Up to date, TJ-II is the only stellarator routinely run on full lithiated walls, thus it offers the possibility to address important issues concerning the possible design of a stellarator-based reactor under very low recycling conditions [1] In this work, the important issues of ion energy confinement, which has been predicted to be strongly modified under pure NBI heating due to minimization of charge exchange loses and the concomitant flattening of edge Ti profiles [2], and central impurity confinement, which has been observed to depend on the type of plasma profile spontaneously developed according to the radial localizations of the source and cooling terms [3], will be particularly stressed. Other topics which have recently been investigated in TJ-II under Li walls are particle retention and release under H/He operation, effect of boron/Li mixtures on sputtering behaviour, edge characteristics of NBI plasmas under peaked, broad and H mode-type profiles, induction of profile changes by particle/impurity injection [3], characteristics of the L-H mode transition and effect of power, density and rational surfaces on the achievement of the transition [4], electric field evolution and its sources, effect of first wall boronization over the Li coating on recycling, impurity generation and transport, etc…

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Edge heat transport in the helical divertor configuration in LHD

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To realize energy supply by fusion reactor, divertor design is one of the most important issues. Understanding of mechanisms determining heat flux profile on divertor plates is necessary for the design. In the Large Helical Device (LHD), the largest heliotron-type superconducting device, particle and heat flux profiles on divertor plates have been investigated by using Langmuir probes and infrared camera. Unlike in the scrape-off layer in tokamaks with poloidal divertor configuration, field lines structure in the edge plasma region in LHD is complicated. There is a stochastic field lines layer outside the last closed flux surface (LCFS) in LHD, and residual islands are embedded in the layer. There are edge surface layers and laminar layers outside the stochastic field lines layer [1], and they connect to divertor plates. Field lines in the stochastic layer connect to divertor plates through the edge surface layer, and their connection length is longer than several hundred meters. On the other hand, connection length of field lines in the laminar layer is typically a few ten meters, and these field lines do not approach LCFS. Therefore, long field lines are main channel of parallel transport of particle and energy from LCFS to divertor, and particles and energy come to laminar layer by perpendicular transport in the edge plasma region. Consequently, particle and heat flux profiles on divertor plates are determined by field lines structure connecting divertor plates and balance of parallel and perpendicular transport. The footprint of field lines on divertor has three-dimensional structure in LHD, and thus, it is observed in experiment that particle and heat flux profiles on divertor plates differ according to location and operational magnetic configuration [2]. Operational conditions, such as density and heating power also affect the profiles in experiment, and it is considered to be caused by changing the balance of parallel and perpendicular transport. Three-dimensional plasma and neutral transport code, EMC3-EIRENE [3] has been applied to study edge transport in LHD [4], and comparisons of particle profiles on divertor plates between experimental observations and numerical results have been conducted [5]. The investigation of heat flux profiles on the divertor plates has been also conducted. In this presentation, mechanisms determining the profiles in LHD are discussed on the basis of this comparison.

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Particle Transport due to Stochastic Magnetic Field in a High-Temperature Plasma

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Direct measurements of the magnetic fluctuation-induced particle flux in the MST reversed field pinch are made using a newly developed differential interferometer in combination with a Faraday rotation system. Measurements show that the magnetic fluctuation-induced electron particle transport (the correlated product of density fluctuations and radial magnetic fluctuations) can account for the total particle flux in the high-temperature plasma core during sawtooth magnetic reconnection events. However, the flux is larger than the ambipolarity-constrained test-particle value predicted for stochastic diffusion. A large ion flux associated with parallel velocity fluctuations correlated with the radial magnetic field fluctuations appears responsible, and indicates a more complete theory for stochastic particle transport is needed.

On Particle Control of the W7-X Divertor

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Recent 3D modelling using EMC3-EIRENE and a previous 2D approach based on B2-EIRENE predict that a high-recycling regime exists for the island divertor of W7-X. In the high-recycling regime, the downstream density increases cubically with increasing separatrix density leading to a rapid reduction of the neutral penetration length towards the core. Associated with the large size of the edge islands in W7-X, the core fuelling rate from the recycling neutrals drops sharply to a level comparable with that predicted for the much smaller W7-AS. This implies that, in the absence of a sufficiently-large inward pinch, achieving high-density plasmas in W7-X would necessitate additional particle central fuelling In this case, the island divertor must be capable of pumping out the externally fuelled particles to maintain the global particle balance, which turns out to be a critical point for the present divertor geometry in W7-X. First, in order to increase the neutral pressure in the divertor chamber, the strike point should be positioned as close to the chamber entrance as possible, which, however, conflicts with power handling. To solve this problem, additional thermal shielding plates are considered and their possible negative consequences on neutral compression have been assessed. Second, the large connection length significantly broadens the particle deposition profile on targets, which on one hand is beneficial, dispersing the power load over the targets, but on the other hand unfavourable for focussing the outflowing ions to increase the neutral influx into the compression chamber. Third, the strike point location as well as the island topology is sensitive to plasma currents induced by finite-ß effects such as the diamagnetic current, the Pfirsch-Schlüter current and, in particular, the Bootstrap current, introducing an additional difficulty in optimizing the particle exhaust condition. An insufficient pumping rate will lead to increase of the SOL density beyond the threshold density of Marfe-formation preventing the core plasma from high-density operation. This paper presents a numerical analysis of these points, aiming at finding optimum particle pumping conditions compatible with energy exhaust in both configuration and plasma parameter space.

Inter-machine edge turbulence data base

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Edge turbulence is responsible for radial transport, which degrades the magnetic plasma confinement in fusion research. A lot of effort has been invested on various devices to investigate turbulence in the plasma boundary. Statistical and correlation properties have been investigated on several machines and by different methods. The poster makes a proposal for an international edge turbulence database to concentrate these efforts and to facilitate inter-machine comparisons and code validation.
Recent investigations on the ASDEX Upgrade tokamak and the TJ-K stellarator revealed clear similarities in the turbulent behaviour of these machines, despite the large differences in magnetic configuration and plasma parameters. This reinforces the relevance of small machines not only for education but also for fusion research and stellarator and tokamak physics.
The data base provides a platform for researchers to easily access and analyse data from various experiments all over the world under the aspect of inter-machine comparison. On the poster, the relevant parameters to be stored are discussed and a data format is proposed. First analysis of data from different devices are presented.

Divertors for quasi-symmetric stellarators

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It is increasingly recognized that very high heat fluxes ($\geq$ 10 MW/m^2) in the divertor regions of magnetic fusion reactors present serious problems for the materials making up the plasma facing components. This has led to the development of a number of flux expansion schemes for tokamaks such as the super-X divertor configuration [1]. The design of divertors for a stellarator is further complicated by the 3-D magnetic geometry, but promising helical and island divertor schemes have been developed. In this paper we begin exploring alternative divertor schemes specifically suited to quasi-symmetric stellarators, using as our first target the quasi-poloidal geometry, for which a variant of the original Spitzer divertor [2] may work without harming the underlying symmetry of the configuration.

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Effects of Toroidal Currents on the Magnetic Configuration of W7-X

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Apart from stiffness of equilibrium, high stability β-limits and low neoclassical transport, the magnetic configuration of Wendelstein 7-X has also been optimized for small bootstrap current. However, this is subject to a balance of toroidicity and helically driven bootstrap current contributions having opposite signs. This balance is influenced by the toroidal mirror field and is well fulfilled in the configuration satisfying all optimization conditions, the so-called high-mirror configuration with a mirror ratio (mr) of about 10% (i.e. |B| varies by ±10% along the magnetic axis). Necessary deviations from this configuration to investigate the confinement properties of W7-X with respect to configuration parameters like rotational transform, mirror ratio or axis position lead to an imbalance. Thus, significant currents can result, although the bootstrap current is, depending on the mirror ratio, still only up to 30% of that of an equivalent circular tokamak. However, a plasma current of 10kA already leads to a change in the value of the boundary-ɩ of 0.0166 at a field strength of 2.5T and a plasma minor radius of a_{eff}=0.5m and major radius R_{maj}=5.5m. For proper island divertor operation, which rests on setting the boundary-ɩ value correctly, a change caused by some 10kA needs to be compensated by some means. Currently, two basic schemes are discussed. First, if the final toroidal net-current is not too large, a proper adjustment of the vacuum ɩ-value with respect to the predicted bootstrap current may provide a good divertor configuration with plasma current. However, as the time-evolution is slow – the L/R-time scale being 20-40s – power deposition zones on the in-vessel components change, while the configuration, starting from a limiter dominated one, evolves to the final divertor configuration. For this scheme, impurity accumulation may become a major challenge. Second, to overcome these long time-scales, a compensation of the bootstrap current with EC-current drive (ECCD) is considered. When this is properly applied, simulations show that one may keep the deviation of the total plasma current in a range of some kA and an almost stationary ɩ-profile can be reached in some seconds. The drawback of this scheme is a strongly deformed internal ɩ-profile due to the mismatch in ECCD and bootstrap current density profiles.

Here, we investigate the above two scenarios and their impact on the equilibrium configuration, including the modification of the internal equilibrium and stability. Low-order rationals introduced by changes in the ɩ-profiles may lead to the appearance of islands for which widths are estimated. An important topic is the change of the boundary/separatrix structure as basis for studies of divertor operation and power load distribution on the in-vessel components. The boundary structure is investigated by extending the vmec2000 [1] equilibria using the EXTENDER-code [2] (similar to vmec/mfbe-code [3]). This approach is compared with HINT2 [4] calculations to investigate the range of configuration changes.

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Magnetic Diagnostics and Plans for Equilibrium Reconstruction at W7-X

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Magnetic loops and probes are basic plasma diagnostics, which are also important for monitoring and controlling discharges, since they are available continuously. Toroidal net-currents and energy content are routinely measured by Rogowski- and diamagnetic coils, respectively. In W7-X, an additional set of segmented Rogowski coils and flux loops will be installed in order to extract more information from the magnetic fields generated by the plasma currents. The complete set is envisaged as the basis for an online equilibrium reconstruction based on Function Parameterization (FP). For this purpose, an already existing dataset of about 10000 vmec2000-equilibria, previously used for an FP to rapidly provide the W7-X equilibrium geometry, has been reused to simulate the responses of the magnetic diagnostics.
This paper gives a description of the magnetic diagnostics foreseen for W7-X and details their different purposes. The planned scheme of equilibrium reconstruction will be described. In this context, the role which the magnetic diagnostics will play will be explained, the status of the analysis of the magnetic diagnostics will be described and its prospects discussed.

Numerical reconstruction of spontaneous helical equilibria in RFX-mod

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The Reversed Field Pinch (RFP) plasma can spontaneously access regimes in which a single saturated resistive kink mode dominates the magnetic perturbation spectrum (Quasi Single Helicity state, QSH) [1]. The helicity of the dominant mode is m=1,n=7 in RFX-mod.
For sufficiently large values of the dominant mode, the magnetic island separatrix is expelled and a helical magnetic axis appears. This becomes the main (and only) magnetic axis of the plasma. These states are the SHAx states (Single Helical Axis); they are characterized by a helically symmetric core plasma, with the periodicity of the dominant mode, within an axisymmetric boundary [2]. The helical character of the configuration is experimentally observed in RFX-mod measuring the electron temperature, electron density and SXR emissivity radial profile. These quantities can be interpreted as flux functions [3] proving a link between the underlying magnetic topology and kinetic plasma quantities. In particular, isobaric surfaces correspond to magnetic flux surfaces.

While a good reconstruction of the magnetic topology in the RFP device RFX-mod was obtained with a perturbative approach in toroidal geometry [4], subsequent study of cross-field transport is considerably complex, since helical flux coordinates need to be determined [5]. Stellarator tools, as the VMEC equilibrium code, can considerably simplify both the determination of perpendicular transport coefficients and the realization of transport simulations in helical geometry.

In this work we present the results obtained with the VMEC code in order to reproduce the magnetic topology of a SHAx plasma in a fixed boundary approximation by providing as input the iota profile, the shape of the last closed flux surface and taking into account also pressure effects. First simulations show a good matching with experimental observations and reconstructions from other numerical codes.

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Remarks on Finite Beta Effects in International Stellarator/Heliotron Scaling

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The study of the international stellarator/heliotron scaling (ISS) based on Intenational Stellarataor/Heliotron Confinement Database is revisited with consideration of finite beta effects for the common understanding of high-beta physics in stellarators and heliotrons. The ISS04, which has been derived from the data of low-beta plasmas [1], shows a large commonality of energy confinement property, whereas the systematic differences due to magnetic configurations such as the amount of helical ripples, ellipticity and so on still remain.

$$\tau_E^{ISS04} = f_{ren}\, a^{2.28} R^{0.64} n_e^{0.54} P^{-0.61} B^{0.84} \iota_{2/3}^{0.41}$$

The parameters are the major radius R (m), the effective minor plasma radius a (m), the volume averaged magnetic field B (T), the volume averaged electron density n_e (10^{20} m^{-3}), the effective heating power P (MW) and the rotational transform $\iota_{2/3}$ (at $r = 2/3a$), respectively. The f_{ren} is a renormalization factor reflecting configuration effects which enables us to extract common gyro-Bohm type feature among different configurations. The beta range is extended to 5% in LHD, and the data shows that global energy confinement gradually degrades with an increment of beta [2]. This degradation is explained by the change of f_{ren} due to a significant Shafranov shift to some extent. Therefore the finite beta effect is a key knob to clarify the configuration dependent parameter. It is predicted that the increment of beta value leads to not only change of the magnetic configuration itself but stochastization of magnetic field structures limiting the confinement region, degradation of the plasma confinement due to MHD instabilities and/or turbulences. In this study, beta dependence of f_{ren} is quantified at first, and the the new scaling including these effects is discussed. The changes of confinement property and equilibrium characteristics with beta value are compared in LHD and W7-AS Stellarator. Especially, beta dependence of plasma confinement property is opposed in both devices. The local transport analyses show that it is caused by the increment of the peripheral transport due to resistive-g turbulence [3] in LHD. The effects of low-order MHD activities on plasma confinement are very weak. The W7-AS high-beta data indicates that the increment of beta gradually improves plasma confinement property, which might be connected with a deepening of the magnetic well and the respective improvement of the MHD stability. This work has been conducted under the Coorinated Working Group Meeting.

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Study of MHD Characteristics by Magnetic Axis Control in high-beta plasmas of LHD

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Magnetic axis position, R_{ax}, is a key parameter characterizing MHD equilibrium, stability, transport and heating efficiency of neutral beams in heliotron configurations. In the FY2008 experiments of the large helical device (LHD), capability of power supplies of poloidal coils determining the R_{ax} was increased to allow a swing of R_{ax} during a discharge, which is valid for the detail optimization of the magnetic configuration for high-beta plasma production, identification of MHD stability boundary, understanding of equilibrium beta-limit and so on. Figure 1 shows a comparison of MHD activities in plasmas with and without R_{ax} swing. The configuration with γ_c of 1.20 and B_t of 0.45 T was applied here. The R_{ax} was preset to change 3.6 m (t = 1 s) to 3.5 m (t = 3 s). The two co-NBI's were applied from 1.3 s and a counter NBI was from 1.8 s in both discharges. The Shafranov shift identified by Thomson scattering system was about less than 20 cm when the averaged beta is about 4 %. Although the temporal changes of central electron temperature were almost similar in both discharges, the rapid drop was observed at 2.2 s in the case of the R_{ax} swing and led by strong m/n = 2/1 mode. Then profile flattening was occurred around the m/n = 2/1 resonance located at $\rho \sim 0.5$. This observation is consistent with a theoretical prediction of ideal MHD stability boundary, and it is due to an enhancement of magnetic hill. The previous experiments in the R_{ax} = 3.5 m configuration indicate that m/n = 2/1 mode limited the achieved beta value [1]. The edge MHD activity with m/n = 2/3 was suppressed by the inward shift of R_{ax}, which is due to a decrease in edge iota leading a reduction of the pressure gradient around the resonance.

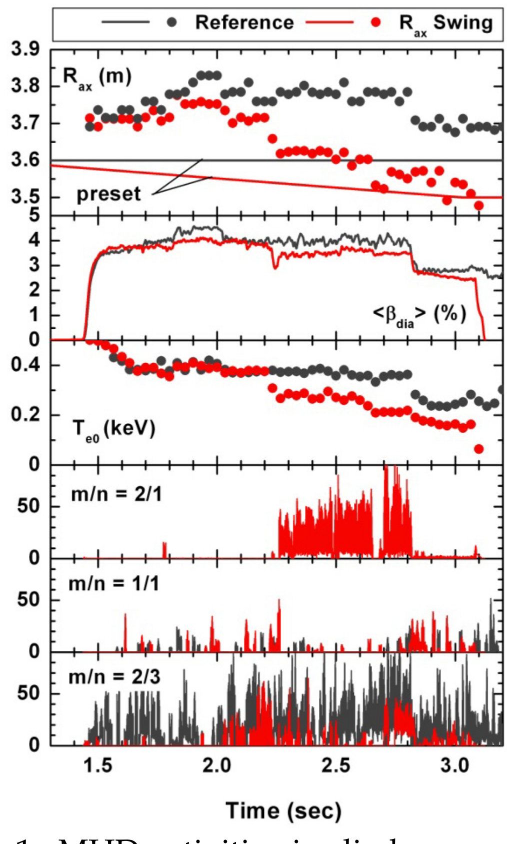

Fig. 1 MHD activities in discharges with and without R_{ax} swing.

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Characteristics of edge MHD modes and ELM activities observed in LHD plasmas

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Control of edge MHD instabilities is one of very important issues to improve plasma performance for stellarator/helical plasmas as well as tokamak plasmas. In high beta L-mode as well as middle beta H-mode plasmas obtained on Large Helical Device (LHD), ideal/resistive interchange modes having low mode numbers, e.g. m/n = 3/4, 2/3 and 1/2 (m, n: poloidal and toroidal mode numbers) which are called “edge MHD modes” are excited noticeably by the formation of steep pressure gradient at the plasma edge, because the edge region has high magnetic shear but in magnetic hill [1-4]. In particular, resonant surfaces related to these edge MHD modes are often located in very edge region at or even outside last closed flux surface (LCFS) defined in the vacuum field. Nonlinear evolution of these edge MHD modes induces repetitive bursts of magnetic fluctuations and Soft X-ray (SX) fluctuations, and generates a train of sharp spikes in Hα emission signals. These edge localized mode (ELM) activities are also observed in high beta or high density L-mode plasmas as well as H-mode plasmas with L-H transition. Resistive interchange modes are thought to be responsible for ELMs in LHD, of which cause of ELMs is clearly differs from that in a tokamak H-mode. In this presentation, detailed spatial and temporal evolutions of edge MHD modes and ELMs are discussed for H-mode and high beta L-mode plasmas on LHD.

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Modification of LHD magnetic configurations based on the boundary shape control for improving confinement properties

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It has been discussed that the stellarator / heliotron concepts of toroidal confinement have a wide undeveloped area of research with new configurations because the three dimensional freedom in the space of configuration design is so wide compared with axisymmetric concepts. As a part of consequences of it, we have several numbers of helical devices working in the world with very different configurations, which is very distinct from tokamak world. Some of these devices were designed (optimized) by tuning magnetic coil structures and some were designed with plasma boundary shape controls.

LHD is the largest one among them and has been producing high quality experimental data with good diagnostic reliabilities and the wide space of plasma parameters (temperature, density and plasma beta). In addition, LHD experiment has actually produced data useful for the configuration optimization work with the control of magnetic axis position and the helical pitch parameter (consequently the magnetic well and the plasma aspect ratio were controlled). Improved data such as high-beta operation was obtained with such configuration control efforts.

The motivation of this work is to extend the magnetic configuration control "virtually" to explore the new configurations which might give better confinement properties than existing data of LHD. Such configurations might be realized by the modification of shapes of LHD coils or truly virtual ones which need completely new coil design. The evaluation of confinement properties should be primarily based on existing LHD experimental data instead of simple traditional theory since the LHD experiment had already exceeded some classical limit of operational boundaries.

The extension of the configuration space is made based on the boundary shape control. The configuration surveys made in the actual LHD experiment was a very limited subspace of such a relatively free design space because they are made with a small number of device control parameters (current control in the existing coils). Better understanding of LHD experimental data for different configurations from the viewpoint of wider configuration space is another purpose of this paper.

Simulation study of MHD stability beta limit in LHD by TASK3D

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The beta limit by the ideal MHD instabilities (so-called "MHD stability beta limit") for helical plasmas is studied for various types of the pressure profiles by a hierarchy integrated code TASK3D. A numerical model for the effect of the MHD instabilities is introduced such that the pressure profile is flattened around the rational surface due to the MHD instabilities. The width of the flattening of the pressure gradient is determined from the width of the eigenmode structure which is evaluated from the linear MHD stability module. It is assumed that there is the upper limit of the mode number of the MHD instabilities which directly affect the pressure gradient. The upper limit of the mode number is determined using a recent high beta experiment in the Large Helical Device (LHD). The flattening of the pressure gradient is calculated by the transport module in the TASK3D. For high beta plasmas, although the interchange mode limits the pressure gradient in the periphery region, the interchange mode is stable in the core region since the magnetic well is generated and its depth becomes deep due to large Shafranov shift in the LHD configuration. In ref.1, we analyzed the achievable beta for two types of pressure profiles. From the analysis, the achievable beta value of 4.2% and 6.0% are obtained for a peaked profile and a broad profile, respectively. The achievable beta values are limited by the equilibrium limit for the peaked profile and by (m,n)=(4,1) mode for the broad profile. In this paper, the simulation study is extended to various types of the pressure profiles and the detail analysis of the beta value will be reported.

[1] M.Sato *et al.*, Proc. of 22nd IAEA Fusion Energy Conf. (Geneva, Switzerland, 2008) IAEA-CN-165/TH/P9-18.

Investigation of equilibrium plasma beta limits in 3-D magnetic topologies

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A fluid model is used to investigate pressure-induced magnetic islands in 3-D equilibria. We revisit previous analytic isolated island calculations while allowing for finite parallel heat transport, to derive an equation for equilibrium island widths. Finite parallel heat transport can alter the impact of resistive interchange and bootstrap current contributions to magnetic island formation. However, the effect of Pfirsch-Schlüter currents driven by resonant components in $\frac{1}{B^2}$ on magnetic island formation is largely unaffected by transport processes.

3D MHD equilibria are modeled using NIMROD. A vacuum equilibrium helical magnetic field is loaded into the geometry of a straight stellarator. The symmetry of the vacuum field with a dominant magnetic harmonic can be spoiled by adding small magnetic perturbations. These perturbations alter the magnetic spectrum, and produce magnetic islands and regions of stochasticity. Numerical simulations are performed that include the effect of a heating source and self-consistent anisotropic transport in a variety of magnetic configurations. The support of pressure gradients in stochastic regions is also investigated.

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Design integration on split and segmented-type helical coils for FFHR

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Based on the progress on fusion relevant plasma experiments in the Large Helical Device (LHD), the conceptual design studies on the heliotron-type fusion energy reactor FFHR are being conducted on both physics and engineering issues. One of the crucial requirements for the physics design is to find sufficient clearances between the ergodic region outside the nested magnetic surfaces and blankets especially at the inboard side of the torus. In order to meet this demand, the currently improved design, FFHR-2m2, has helical coils with fairly large major and minor radius of R_c ~17 m and a_c ~4 m, respectively. On the other hand, it has been found as an alternative design that equivalent clearances are obtained by employing a weaker twisting of helical coils together by splitting them in the poloidal cross-section at the outboard side [2]. This configuration, named FFHR-2S Type-I, has R_c = 15 m and a_c = 3 m. On the other hand, split-type helical coils are found to provide another configuration, FFHR-2S Type-II, which ensures magnetic well formation in the fairly large nested magnetic surfaces (shown in Fig. 1) with an outward shift of the magnetic axis at the size of FFHR-2m2. Various physics properties are being examined for these configurations, such as drift orbits of high-energy particles. From the engineering viewpoint, we propose that such complicated helical coils (of split type) and huge size (also for non-split type) be constructed by prefabricating half-pitch segments using high-temperature superconductors (HTS); the segments are then to be assembled on site with a number of joints [2, 3]. Short-sample tests with reduced-scale YBCO conductors are validating the technical feasibility. Related issues for realizing the HTS option are also being investigated, such as the error magnetic field created by shielding currents in non-twisted HTS tapes, ac losses, mechanical strength, bending strain during the winding process and so on.

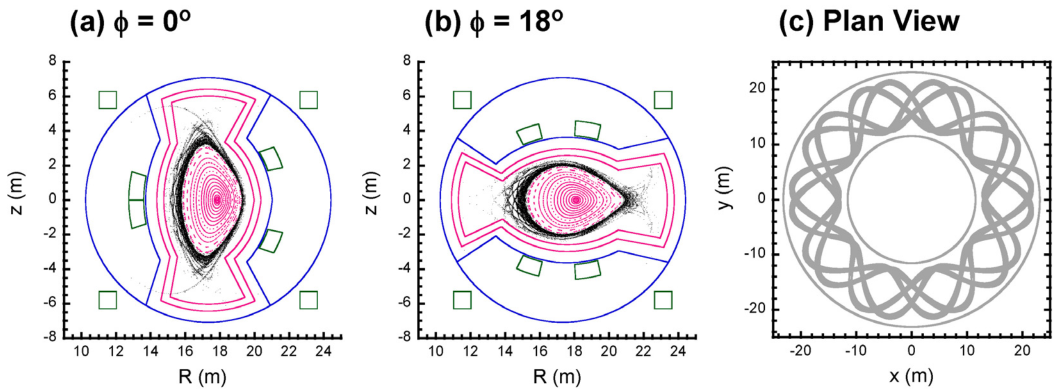

Fig. 1 Vacuum magnetic surfaces and a plan view of the coils of FFHR-2S Type II.

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[3] Bansal, G. et al., Plasma Fusion Res. **3** (2008) S1049.

IMPROVED W7X CURRENT LEAD TEMPORARY SUPPORT AND ASSEMBLY ARRANGEMENT*

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A new design approach has been defined for the installation of the W7X current leads (CL) which simplifies the assembly procedure and reduces the assembly time. An intermediate gravity support system is used to support the CL off an upper dome vacuum structure allowing individual coil leads to be installed and a final, independent installation of the lower current lead vacuum structure. The initial baseline approach required the installation of a pair of current leads concurrently and an involved manipulation and temporary support of the vacuum shell structures. The revised CL installation approach includes a four-point temporary support system and a modification of the upper cryostat CL dome structure at four locations. One critical aspect of the new support arrangement is the ability to access temporary support pins that are installed in the confined regions, surround by port components. A visual check of the CAD model with all ports installed indicates that pin access is acceptable. To further validate the design, CAD models of the W7X cryostat, all relevant local ports, the current lead support system and the current lead cryostat dome structure was imported into a 3D stereo display computer system to allow a virtual simulation of installation area. The design details of the new CL support and installation system along with the results of the 3D assembly simulation will be presented.

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Configuration Space Control for Wendelstein 7-X

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The Wendelstein 7-X stellarator (W7-X) is a superconducting fusion experiment presently under construction at the Greifswald branch of the Max-Planck-Institut für Plasmaphysik in Greifswald, Germany. W7-X is a device with extreme geometrical complexity due to the close packing of components inside the cryostat and their complex three-dimensional shapes. The task of the Configuration Space Control department is to ensure that these components do not collide with each other under the defined set of configurations such as during assembly, at cool down, or during operation at various coil currents, among others. To fulfill this task, sophisticated tools and procedures were developed and implemented within the realm of a newly founded division that focuses on design, configuration control, and configuration management. This paper will discuss the Configuration Space Control process, explore the advantages to the project resulting from the process, and demonstrate its application in the analysis of the cryogenic cooling pipes of Module 5.

Energetic-Electron-Driven Instability in the Helically Symmetric Experiment

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Energetic electrons generated by electron cyclotron resonance heating are observed to drive instabilities in the quasi-helically symmetric HSX device. The coherent, global fluctuations peak in the plasma core and are measured in the frequency range of 20-120 kHz. Measurements show this mode can be continuous or bursting, and in some cases have small frequency chirps. Mode propagation is in the diamagnetic drift direction of the driving species. Measured mode helicity is toroidal mode number and poloidal mode number (and odd). Multiple modes are observed with mode spacing ~22 kHz. The very weak dependence of the mode frequency on inverse safety factor ι distinguishes the observed mode from shear Alfvénic perturbations. Acoustic modes are the favored global modes in the shear Alfvén-sound wave spectrum of low shear devices like HSX as they are weakly damped and therefore susceptible to energetic particle destabilization. Mode amplitude sensitivity to magnetic ripple also supports the likelihood of observed instability being a sound wave.

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Modeling Fast Ion Transport in TAE Avalanches in NSTX

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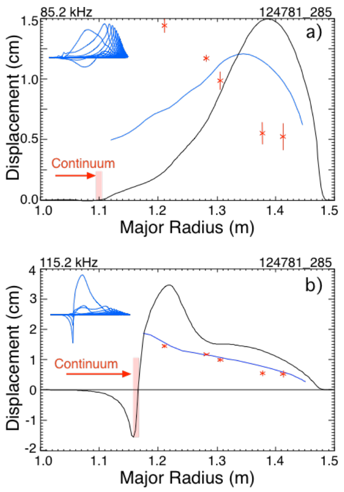
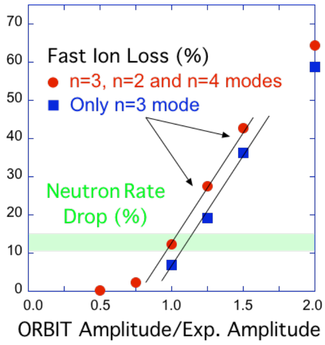


While single TAE cause negligible fast ion transport in NSTX, multiple, strongly interacting modes as in an "avalanche event", can cause substantial fast ion loss. Interaction of multiple modes on ITER in a similar manner could redistribute or cause the loss of fusion αs.

Detailed measurements of the equilibrium, mode structure and amplitude are used to model the effect of the multiple TAE on fast ion confinement. The NOVA code is used to calculate the shape of the TAE eigenfunctions. For the experimentally measured equilibrium, multiple eigenmodes are found. In Figure 1 the shapes of the NOVA eigenfunctions (blue inset and the solid black line shows the displacement on the outboard midplane) are compared with measurements from an array of five reflectometers (red points). The blue curve is the simulated reflectometer response showing a good fit to the lower mode shape.

The NOVA eigenfunction which best fits the reflectometer data is used in the ORBIT code to simulate the enhanced fast ion transport. The measured amplitude and frequency evolutions for a one ms avalanche burst are used to scale the linear NOVA eigenmodes shapes in an ORBIT simulation. The unperturbed fast ion distribution is calculated with the TRANSP Monte Carlo beam deposition code and used as input to ORBIT. Simulations find reasonable qualitative agreement between the predicted fast ion losses and estimated experimental the fast ion losses. The simulations do predict enhanced fast ion transport for a wide range of energies, as seen experimentally. Also, the ORBIT simulations suggest that the strong frequency chirping also plays an important role in fast ion redistribution.



Work supported by U.S. DOE Contracts DE-AC02-09CH11466, DE-FG03-99ER54527, DE-FG02-06ER54867, and DE-FG02-99ER54527.

Heating Position Dependence of Energy Spectra of Fast Ions Generated by ICRF Heating in Heliotron J

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Fast ion velocity distribution is investigated using fast protons generated by ICRF minority heating in Heliotron J, a low-shear helical-axis heliotron (R_0 = 1.2 m, a = 0.1-0.2 m, $B_0 \leq$ 1.5 T), with special emphasis on the effect of the toroidal ripple of magnetic field strength and heating position. The configurations used in this study are as follows; the bumpiness (B_{04}/B_{00}, where B_{04} is the bumpy component and B_{00} is the averaged magnetic field strength) are 0.15 (high) and 0.06 (medium) at the normalized radius of 0.67. The configuration of B_{04}/B_{00} = 0.06 corresponds to the standard configuration in Heliotron J. The fast ions are measured by a charge-exchange neutral particle energy analyzer (CX-NPA) installed at the opposite position in the toroidal angle to the ICRF antennas. From the measurement, the pitch angle dependence of the energy spectra for three bumpy configurations is observed. The majority species of plasma is deuteron and the minor is proton. The minority ratio is about 10%.

The ICRF wave of the frequency of 23.2 MHz for the high bumpy case and 19 MHz for the medium and low bumpy cases is injected into ECH plasmas in the central heating condition. The magnetic field and the ICRF frequency are selected so that the ECH resonance is positioned at the axis of the plasma. The magnetic field is 1.25T at the plasma axis in the ECH injection section. The line-averaged and ICRF injection power are 0.4 x 10^{19} m^{-3} and about 0.3 MW, respectively. The high energy component is observed near the pitch angle of 120 deg in the range of observation from 111 deg to 128 deg in the high bumpy case. The tail component extended to about 30 keV is observed only in the high bumpy case [1].

The heating position is changed by ICRF frequency in the medium bumpy case. The frequency of 19 MHz is used for the on-axis heating and 23.2 MHz is for the inner-side off-axis heating. The high energy tail of minority ion is larger in the on-axis heating whereas the bulk ion temperature is higher in the off-axis heating. In the minority heating, the bulk heating is mainly due to Coulomb collisions with fast ions accelerated by ICRF heating. It is supposed that fast ions are localized in the toroidal or poloidal direction, or in the velocity space from the complex loss cone structure of Heliotron J field. The fast ion spectra in the opposite side of the pitch angle of 90 deg are found to be asymmetric against 90 deg in the field reversal experiment. A Monte-Carlo analysis of the fast ion behaviour is also in progress.

[1] H. Okada, et al., “Velocity Distribution of Fast Ions Generated by ICRF Heating in Heliotron J”, Proc. 22nd IAEA Fusion Energy Conference (2008) EX/P6-28.

Collective Thomson Scattering of 77 GHz High Power ECRH Beam in LHD

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The measurement of the ion velocity distribution is especially important in reactor relevant plasmas. The collective Thomson scattering (CTS) is one of the most promising methods for evaluating the ion velocity distribution function. In spite of its potential, this method had long been suffered from the absence of adequate power sources. Recent development in the higher power, and higher frequency range of the gyrotron and the transmission technique enabled to realize the measurement of not only the bulk but also high energy component of the ion velocity distribution function.
One of the newly installed 77 GHz gyrotrons in LHD can generate more than 1MW over 5 s. This gyrotron is connected to one of the Gaussian mirror antenna set. This antenna set includes one another Gaussian beam mirror suitable for receiving the scattered power from definite scattering volume.
In Fig.1 are shown the plasma and the beam cross section on the vertically elongated section in LHD. For use of this gyrotron, operational magnetic field should be chosen near B_0=2.2 T on axis so as to exclude the fundamental and second harmonic resonances of 77 GHz on the line of sight inside the plasma confinement region, in order to reduce the ECE background that is considered to be the largest noise source. A heterodyne receiver system of a fundamental mixer with fixed local oscillator is was installed on the upstream of the transmission line. A notch filter with the 3 dB band width of 300 MHz was used for the front end. The adjustment and trial of the CTS using above mentioned system was done during the last experimental campaign of LHD. Signals that can be attributed to the CTS were obtained, although the magnetic field setting was 2.75 T where the fundamental resonance lies close to the magnetic axis as also shown in Fig.1 Analysis results of these preliminary results and the improvements of the receiver system for the coming experimental campaign are discussed.

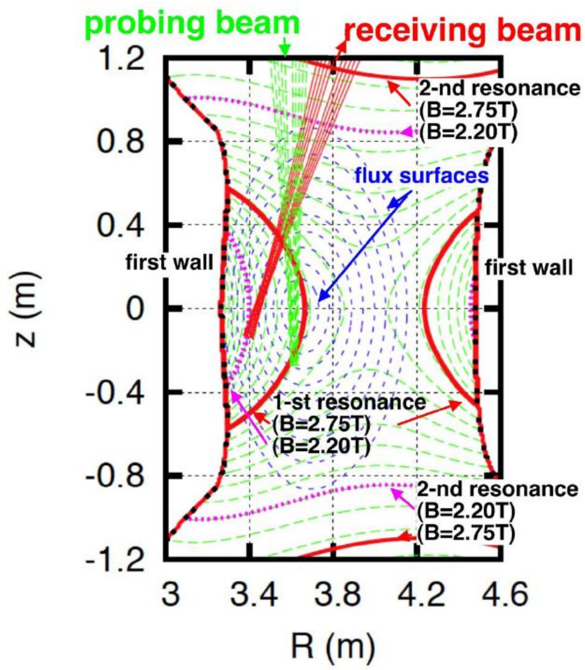

Figure 1: Relation between probing, receiving beams, flux surfaces and EC resonances for the cases of magnetic field setting B_0=2.2 and 2.75 Tesla on the vertically elongated cross section in LHD.

Monte-Carlo Study of Perpendicularly Injected High-Energy Particles using Real Coordinate System in LHD

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A perpendicular neutral beam (NB) injector as well as three tangential-NB injectors has been installed on LHD. With the use of these NBs, a volume averaged beta has been reached 5% in the recent LHD experiments[1]. In order to maintain such high beta plasmas, it is one of the important issues to investigate the pressure due to the high-energy particles produced by the NBs. The beam-pressure can be estimated with the use of the Monte-Carlo simulation based on the high-energy particle tracing. The conventional Monte-Carlo simulation studies of LHD, in which the Boozer coordinates is used, have been made only for the region within the last closed flux surface (LCFS)[2]. In the high beta plasmas of LHD, the flux surfaces in the periphery are destroyed and, as a result, the volume of the LCFS becomes small. Additionally, the magnetic axis shifts in a direction of the major radius because of the Shafranov shift. It is pointed out that the re-entering particles[3], which repeatedly pass into and out of the LCFS, play an important role for the high-energy particle behavior in such magnetic configurations. Besides, the particles produced by the perpendicular-NB tend to be re-entering particles[3]. In the conventional studies, however, re-entering particles are regarded as the lost particles. Therefore, the beam-pressure cannot be obtained accurately. We have developed a new Monte-Carlo code based on the orbit following Monte-Carlo code[4]. The new code allows for the calculation of the drift orbits with high accuracy in a complex magnetic field configuration solving the guiding-center equations in the real coordinates with the use of the equilibrium magnetic field calculated by the HINT[5]. In this code, the re-entering particles are also traced appropriately since the real coordinates is used. The collision effects are taken into account using the Monte-Carlo collision operator[6]. Furthermore, the particle loss due to the charge exchange reaction outside the LCFS is included[7]. In order to investigate the effect of the re-entering particles on the pressure, the developed code has been applied to the high-energy particles produced by the perpendicular-NB in LHD. Especially, the particles in the high beta plasmas have been studied in detail. In the conference, the outline of the developed code will be shown. The distribution function and the pressure of the high-energy particles produced by the perpendicular-NB injected in LHD will be presented. The effect of the re-entering particles on the distribution function and the pressure will be discussed.

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Improving the tokamak and RFP concepts by addition of 3D fields (and use of stellarator tools for their study)

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Non axisymmetric perturbing fields can provide a variety of beneficial effects in tokamaks. ELM mitigation and RWM feedback stabilization are familiar examples. As well, there is the possibility of using 3D perturbing coils to allow start-up without transformer assist, and to passively stabilize the vertical stability of tokamaks. We will describe analytical and numerical calculations which investigate the potential for realizing these benefits using outboard stellarator windings.
Single helical axis (SHAx) states have been observed in RFP plasmas in RFX. Compared with conventional multiple helicity RFP plasmas, SHAx configurations exhibit much improved transport. It has been suggested that sustained SHAx configurations can be induced in RFX by imposing helical fields using the extensive saddle coil system surrounding the vacuum vessel. A brief description of calculations which examine the saddle coil current requirements will be given (combining VMEC and NESCOIL) as well as a description of a novel use of VMEC to produce helical axis configurations.

Non-Axisymmetric Helical Structures in the MST Reversed-Field Pinch

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Reversed-field pinch (RFP) particle confinement is degraded by stochastic magnetic fields stemming from multiple resistive tearing modes that produce overlapping islands. RFP simulations reveal an alternate state where the normally broad spectrum of tearing modes condenses into a single, large helical mode. These two states are referred to as multiple helicity (MH) and single helicity (SH), respectively. In the SH scenario, the plasma is expected to have closed helical flux surfaces within the helical structure and can be considered analogous to the non-axisymmetric stellarator equilibrium. The term quasi-single helicity (QSH) has been adopted to describe plasmas where there exists a dominant mode and additional secondary modes of lower amplitude. QSH has led to a substantial global confinement improvement in the RFX-mod RFP. Studies have been carried out in the Madison Symmetric Torus (MST) on both MH and QSH plasmas in order to investigate differences between these states. For the first time, the internal magnetic topology as well as magnetic and density fluctuations associated with these modes were measured directly in the plasma core using a high-speed, laser-based, polarimetry-interferometry diagnostic. The density and radial magnetic field fluctuations have been correlated in order to evaluate the magnetic-fluctuation-induced particle flux in the high-temperature core region. The electron temperature profile is 50 ev hotter for $0< r/a <0.75$ at the O-point in QSH plasmas, as opposed to the X-point. Initial analysis of the soft x-ray tomographic reconstuctions depicts an island of high emissivity correlated with a denser region seen in the line-integrated density. Previous hard-x-ray measurements indicate improved localized particle confinement concurrent with the hot island formation. Quantitative comparisons of core and edge fluctuation measurements as well as global particle confinement are underway to identify properties of the QSH and MH states.

Overview of the DIII-D non-axisymmetric center post coil design and physics basis*

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A new small aperture, high-field side, non-axisymmetric magnetic perturbation coil is being installed on the DIII-D center post (CP)-coil. This coil has been designed to study the effects of resonant and non-resonant 3D magnetic field perturbations on plasma confinement and stability. As shown in Fig. 1, the design provides a significant enhancement in our ability to control the radial profile of the resonant magnetic field spectrum compared to that of the existing moderate aperture, low-field side, I-coil and the large aperture, low-field side, C-coil. By combining the fields from each of these 3 coils, we can create localized peaks in the resonant profile and position these peaks at various points across the pedestal giving us better control over the degree of island overlap and open field line loss fraction. The CP-coil is comprised of 3 vertical rows of coils, centered on the DIII-D equatorial plane, with 12 rectangular conducing loops per row, which provides a significantly wider range of variation in the ratio of resonant to non-resonant field components than either the I- or C-coil. The CP-coil also provides increased flexibility for specifying the 3D distribution of the non-resonant fields with respect to intrinsic field-errors and diagnostic systems that are fixed in space due to the increased toroidal phase angle resolution provided by the 3×12 coil array vs the 2×6 I-coil and the 1×6 C-coil array. In addition, the CP-coil is designed to produce fast 3D field pulses with full current rise times of 3 ms for high-time resolution studies of changes in the properties of the pedestal turbulence and pedestal profiles as a function of coil and plasma parameters. Initially, the CP-coil will be configured for n=3 operations to connect with edge localized mode suppression results from the low-field side non-axisymmetric coils and to address physics issues that are not accessible with either the I-coil or the C-coil. For example, the CP-coil can be programmed to produce the same $|\delta b_\perp|$ on a particular q=m/n resonant surface in the pedestal as the I-coil but with a much smaller $|\delta b_\perp|$ on resonant surfaces in the core (Fig. 1). This allows us to produce an I-coil similar vacuum stochastic layer across the pedestal with the CP-coil that has significantly smaller core resonant islands. An overview of the new physics capabilities provided by the CP-coil and a summary of the near-term experimental program planned for the combined 3D coil system on DIII-D, will be presented.

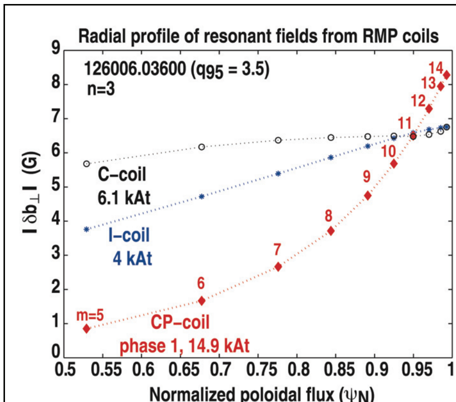

Fig. 1 Radial profile of resonant n=3 fields from each of the DIII-D non-axisymmetric coils.

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Transition to helical RFP state and associated change in magnetic stochasticity in a low-aspect-ratio RFP

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Abstract
Recent progress in RFP research has demonstrated that an RFP state where a single helical mode dominates the plasma dynamics can lead to improved confinement, as Quasi-Single Helicity (QSH) or Single Helical Axis (SHAx) state in RFX. The confinement improvement is attributed to the recovery of magnetic flux surfaces (magnetic chaos healing) in otherwise stochastic core region. In fact, it has been shown that the electron temperature profile has much steeper gradient in the core region in QSH or SHAx RFP states in RFX. Similar steeper temperature gradient has also been observed in MST RFP where improved confinement has been achieved by current profile control using the pulsed poloidal current drive (PPCD) for tearing mode stabilization.
In a small low-aspect-ratio RFP machine RELAX (R/a = 0.5m/0.25m, I_p <100kA)[1], we have observed easier transition of the magnetic configuration to QSH whose duration might depend on characteristic time of dissipation in standard low-A RFP plasmas[2]. Moreover, the dependence of the m=1 magnetic fluctuation amplitudes on the field reversal parameter F ($=B_\varphi(a)/\langle B_\varphi \rangle$, the ratio of the edge toroidal field to the average toroidal field) has shown that the amplitudes deviate largely when the RFP discharge is operated near the boundary of the field reversal, F~0. In high amplitude region, radial profile measurements of the magnetic fields (B_r, B_θ, B_φ) with an array of magnetic probes inserted to the magnetic axis has revealed transition to helical RFP state[3]; The profiles of both the axisymmetric and m=1 asymmetric (fluctuating) components have shown good agreement with those predicted by the Helical Ohmic Equilibrium state, where m=1 helical RFP configuration can be sustained Ohmically without magnetic stochasticity. In RELAX, however, the amplitudes of the neighboring modes are not negligible in the observed helical RFP state. Magnetic stochasticity both in QSH and helical RFP states has been examined by running a numerical code for field line tracing using the measured mode spectrum. The complexity in low-A RFP configuration would be reconstruction of the eigenfunctions of the tearing modes, and the resistive wall boundary condition in RELAX. Effects of the aspect ratio and magnetic boundary conditions on the transition to QSH, further transition to helical RFP and associated change in magnetic stochasticity will be discussed.

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NCSX CONSTRUCTION ACCOMPLISHMENTS AND LESSONS LEARNED*

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The National Compact Stellarator Experiment (NCSX) was designed to test a compact, quasi-axisymmetric stellarator configuration. Flexibility and accurate realization of its complex 3D geometry were key requirements affecting the design and construction. Limiting the field errors that can cause magnetic islands in the plasma was a requirement that was both important and challenging. State-of-the-art 3D CAD modeling and a range of metrology tools were successfully used to address these problems. Field error control measures include accurate construction (e.g., ±1.5 mm tolerance on the coil current center position) and the use of trim coils for field error compensation. While the project was terminated before completing construction, there were significant accomplishments in design, fabrication, and assembly. The design of the stellarator core device was completed. All of the modular coils, toroidal field coils, and vacuum vessel sectors were fabricated. Critical assembly steps were demonstrated, including the assembly of two (of six) three-coil modules to tolerance specifications, and the successful trial installation of a one module over a vacuum vessel sector. Engineering advances were made in the application of CAD modeling, structural analysis, and accurate fabrication of complex-shaped components and sub-assemblies.

There were many lessons learned from the NCSX experience, both technical and managerial, and both general and stellarator-specific. An analysis of completed work packages showed that tight tolerances and complex geometries were about of equal importance as underlying factors affecting cost and schedule growth. Reviews of project management history on the project highlighted the importance of having adequate engineering staff in all phases, completing sufficient design and R&D before budgets are established, and having a formal risk management program. This paper will summarize both the technical accomplishments and the lessons learned.

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Core Plasma Design of a Heliotron Reactor

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The design of core plasma with the LHD-type heliotron configuration has been advanced. One of the most critical problems in the design of LHD-type heliotron reactor is the simultaneous achievement of the substantial plasma volume that enables a good confinement property and the sufficient space for the blanket and the shielding. One solution is an enlargement of the reactor size, but the specification of superconducting components (e.g., coil current density, stored magnetic energy, etc.) can be limited from a technological standpoint. It was discovered that not only the geometry of helical coils but also the placement of poloidal (vertical field) coils has a considerable influence over the equilibrium magnetic surface structure including the stochastic layer. Then the placement of poloidal coils has been examined in detail to enlarge the volume enclosed by the last closed flux surface and to avoid interference between plasma and the surrounding engineering components (Fig. 1). We also have developed a system design code for heliotron reactors, which can deal with a wide design space. The parametric scan with this system code indicated that the volume-average beta value of >5% is required to maintain self-ignition plasma. The existence of this high-beta plasma equilibrium has also been confirmed by the numerical analysis with the 3-D equilibrium code VMEC. This method enables the design of fusion core plasma with heliotron systems that is consistent with the reactor engineering design.

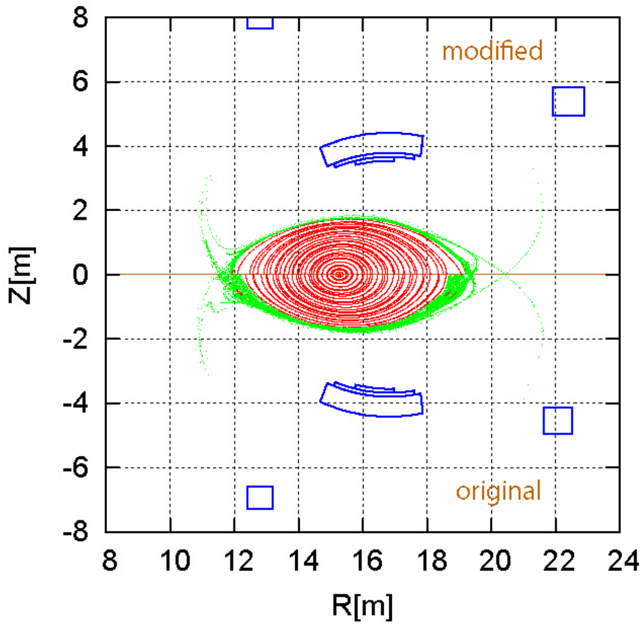

Fig. 1: comparison of magnetic surface structure
with original (lower) / modified (upper) poloidal coil position

Extrapolation of the W7-X magnet system to reactor size

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The fusion experiment Wendelstein 7-X (W7-X) which is presently under construction at the Greifswald branch institute of IPP shall demonstrate the reactor potential of a HELIAS-type stellarator. The magnet system is built up of five identical modules where each one consists of two flip-symmetrical half-modules encompassing 5 non-planar and 2 planar superconducting coils of different geometries. All ten coils of a type are connected electrically in series by superconducting bus-bars and energized by a separate power supply. The non-planar coils produce the stellarator field proper, whereas the planar ones allow additional field variations and are intended to enhance the experimental flexibility. A W7-X – type HELIAS fusion reactor (HSR) would contain only the 50 non-planar coils.

Such reactors are under study at IPP since many years. Besides the fivefold symmetry as in W7-X, three- and four-periodic versions are investigated too. In all cases the reactor coils evolve from those of W7-X by linear scaling of the main dimensions by about a factor of four. This yields dimensions which come close to those of the ITER toroidal field coils, and by coincidence the lengths of the winding pack centre lines are ≈34 m and agree within 2 %. In the most recent HSR versions the characteristics of advanced superconductors like the latest developments of Nb_3Sn and Nb_3Al are taken into account. State of the art Nb_3Sn already allows to increase the maximal conductor induction from the previously assumed 10 T up to at least 12 T as in ITER, corresponding to 5.5 T at the plasma axis. However, non-planar coil construction seems to be quite challenging due to the brittleness and strain sensitivity of Nb_3Sn. The alternative Nb_3Al would be very attractive due to its superior superconductor characteristics, in particular concerning strain persistency which would significantly facilitate the coil fabrication. Nb_3Al has by now proceeded to a highly advanced state of development and is at the brink of industrial production; it can be considered as a serious candidate for a future stellarator reactor with potentially 13 T and more at the coils. The maximal field would practically not be limited by the superconductor but rather by structural integrity.

The new code "MODUCO", based on Bézier curve approximations, was created for interactive coil layout. This tool which can also compute magnetic surfaces and forces was used to design an HSR5 reactor with conservative 12 T maximal induction at the windings. Its total coil current turns out to be 13.4 MAt and is thus larger as compared to the corresponding ITER-value of 9.1 MAt. Using Nb_3Al, the cross section of the superconductor proper would still stay smaller than within an ITER winding pack. The centripetal forces acting on such a HSR5-coil are about a factor of 2.5 lower than in ITER, the local maximal forces per coil unit length are ≈20 % higher. For an updated HSR5 conceptual magnet design one should therefore take advantage of this comparability of the coil sizes and structure requirements and try to transfer the well developed ITER-technologies wherever possible.

This last HSR5-version as well as the basic ideas of MODUCO will be presented. In addition, first structural FE analysis results, superconductor options, and possibilities to use ITER design solutions for the magnet system of such a stellarator reactor will be discussed.

QAS DESIGN OF THE DEMO REACTOR

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The NSTAB code solves differential equations in conservation form, and the TRAN test particle code tracks guiding center orbits in a fixed background, to provide simulations of equilibrium, stability and tranport in tokamaks and stellarators. These codes are well correlated with experimental obsevations and have been validated by convergence studies. Bifurcated 3D solutions of the 2D tokamak problem have been calculated that suggest persistent disruptions and ELMs crashes will occur in ITER. Therefore we have designed a QAS stellarator with similar proportions as a candidate for the DEMO fusion reactor [1]. Our configuration has two field periods and an exceptionally accurate 2D symmetry that furnishes excellent thermal confinement and good control of the prompt loss of alpha particles. Robust coils are found from a filtered form of the Biot-Savart law based on a distribution of current over a control surface for the coils and the current in the plasma defined by the equilibrium calculation. Computational science has settled these harder issues of mathematical physics, so what remains to be developed is an effective plan to construct the coils and build a divertor.

[1] P.R. Garabedian and G.B. McFadden, J. Research NIST, 2009.

STELLARATOR CONFIGURATION IMPROVEMENT USING HIGH TEMPERATURE SUPERCONDUCTING MONOLITHS*

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Substantial advances have been made in the design of stellarator configurations to satisfy physics properties and fabrication feasibility requirements for experimental devices. However, reactors will require further advances in configuration design, in particular with regard to maintenance and operational characteristics, in order to have high availability. The diamagnetic properties of bulk high temperature superconductor (HTS) material can be used to provide simple mechanisms for magnetic field-shaping by arranging them appropriately in an ambient field produced by relatively simple coils.

A stellarator configuration has been developed based on this concept, using the ARIES-CS design point and component features as a point of departure. A small number of toroidal field coils is sufficient to create the background toroidal field. Discrete HTS monoliths ("pucks" or "tiles") are placed on a shaped structure that can be split in the poloidal direction at arbitrary locations. This allows the stellarator to be designed with large openings that provide access to remove interior plasma facing components, no longer restricted by highly shaped back legs of the modular coil winding. Unlike a coil, the structure can be assembled and disassembled in pieces of convenient size, facilitating maintenance.

The excellent properties of HTS materials, e.g., YBCO operating at elevated temperatures (> 30 K), also offer operational advantages. Since the HTS monoliths require no insulation or copper for stability/quench protection, some of the typical irradiation limits on these materials are eliminated. Nuclear heating, due to the high temperature of operation of the HTS compounds, is also very much relaxed, since at 50 K it is possible to remove more than one order of magnitude higher cryogenic loads than at 4 K, for the same refrigerator power. At the same time, there are challenging issues, such as mechanical support and cooling of the monoliths, performance and lifetime limitations in the fusion environment, field creep, superconducting stability of the monoliths, and cryostat design.

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Thursday

Oct. 15, 2009

08:30 – 09:15	PL03	M. Hirsch	***Overview of LH-transition experiments in helical devices***
09:15 – 09:45	I23	T. Estrada	***L-H transition experiments in TJ-II***
09:45 – 10:15	I24	F. Sano	***Study of Improved Confinement Modes in Heliotron J(***
10:15 – 10:45	I25	P. Xanthopoulos	*Gyrokinetic microturbulence investigations towards anomalous optimization*

Coffee

11:15 – 11:45	I26	M. Ramisch	*Investigation of turbulent transport and shear ows in the edge of fusion plasmas*
11:45 – 12:15	I27	M.A. Pedrosa	*Long-range correlations during plasma transitions in the TJ-II stellarator*
12:15 – 12:45	I28	N. Tamura	*Characteristics of nonlocally-coupled transition of the heat transport in LHD*

Lunch

Excursion

Overview of LH-transition experiments in helical devices

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LH transition phenomena, the rapid switch of turbulent transport at the plasma edge followed by the development of an edge pedestal and ELMs have been found in Stellarator-, Heliotron- and Heliac configurations as well, confirming the generic character of the underlying plasma edge physics. For the classical 2D tokamak H-mode the paradigm of a bifurcation in the system of flows and turbulent transport-carrying vortices is widely accepted albeit so far it has not consistently been proven by modelling or experiments.

Dedicated experiments on the nature of the LH-transition itself require highly time resolving diagnostics probing the pedestal region. Where available the results gained at helical devices confirm a close and immediate correlation between the three elements velocity shear, turbulence suppression and increase of edge profile gradients in agreement with the classical paradigm of turbulence suppression by sheared flows.

In helical devices the comparatively uniform picture established for the classical 2D tokamak H-mode is supplemented by elements particular for 3D. They offer not only an opportunity for improved understanding of the underlying bifurcation physics but also yield potential control parameters in view of a reactor relevant scenario:

The influence of the magnetic configuration - its edge topology, the presence of islands and the degree of ergodization - is obvious in the experiments, but the underlying physics such as adding damping and driving elements to the generation of flow shear is not yet clearly understood.

In 3D ambipolarity is not intrinsically ensured and the radial electric field being non-linear dependent already on the neoclassical fluxes results in bifurcation phenomena and an $E \times B$ flow shear even prior to a LH-transition. This negative E_r of the ion root equilibrium conditions seems to play the role of a biasing element facilitating the further dynamic spin-up of rotation in the H-mode. In W7-AS for example the so called Optimum Confinement state with large flow shear and even ELM-like phenomena developed under strong ion-root conditions but did not show a sudden transition or back transition as observed for the H-mode. The control element besides the magnetic configuration was the edge density profile.

A further but now open issue is how turbulent transport and flows and thus the LH-transition are influenced by the elements of stellarator optimization which have been developed primarily to optimize the equilibrium and reduce the convective fluxes. It may be expected that the quasi-symmetries of the magnetic field strength on a flux surface and the minimization of their geodesic curvature not only improve neoclassical transport but also influence preferred flow direction and coupling between flows and turbulence and thus the elements of the LH transition.

L-H transition experiments in TJ-II

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In the TJ-II stellarator, spontaneous L-H mode transitions are achieved [1] under NBI heating conditions when operating under lithium coated walls [2]. H-mode transitions reproduce common features found in other devices [3]: i.e. an increase in plasma density and plasma energy content, a reduction in H_α signal and the development of steep density gradients. Besides, measurements carried out by Doppler reflectometry [4] show an increase in the negative radial electric field and radial electric field shear, and a significant reduction in the level of broadband fluctuations [5]. The transitions are observed at moderate input powers, with one (co-injector) and with two (co- and counter-) NB injectors (about 500 kW port through, each).

Several magnetic configurations have been explored in order to study the influence of the rotational transform in the transition. The shear of the negative radial electric field increases at the L-H transition by an amount that depends on the magnetic configuration and NBI heating power. Magnetic configurations with and without a low order rational surface close to the plasma edge show differences that may be interpreted in terms of local changes in the radial electric field induced by the rational surface that could facilitate the L-H transition.

Fluctuation measurements show a reduction in the turbulence level that is strongest at the position of maximum radial electric field shear. High temporal and spatial resolution measurements indicate that turbulence reduction precedes the increase in the mean sheared flow, but is simultaneous with the increase in the low frequency oscillating sheared flow. These observations may be interpreted in terms of turbulence suppression by oscillating sheared flows.

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Study of Improved Confinement Modes in Heliotron J

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One of the major objectives of the Heliotron J project is to explore a concept optimization for the helical-axis heliotron. In Heliotron J experiments, we have observed improved confinement modes under several experimental conditions. This paper reviews the peculiarities of these improved modes and discusses scenarios to obtain a higher (or more preferable) confinement state.

Studies of improved confinement modes accompanied by spontaneous changes of macroscopic parameters in the edge region have revealed the threshold density for the transition and the existence of magnetic configuration dependence of the “H-mode quality”. The “high-quality H-mode” is located in the specific vacuum iota range of slightly less than the values of the major natural resonances [1]. Co/CTR NBI experiments suggest the importance of iota change caused by non-inductive plasma current and/or the direction of momentum input [2]. It is suggested that the reduction of the fluctuation-induced transport in the SOL and the formation of the negative E_r (or E_r-shear) near the LCFS at the transition. It was also observed that the apparent motion of a filamentary structure in a peripheral plasma fluctuation changed its direction in the H-mode, suggesting build-up of negative E_r in the H-mode [3].

The effects of the bumpiness on the global confinement have been investigated selecting three configuration with different bumpiness [4, 5]. The confinement in the high and medium bumpy configurations is better than that in the low bumpy configuration. The different bumpiness dependence for the ECH and NBI plasmas has been observed, possibly caused by the improved fast-ion confinement for the NBI plasmas in the high bumpiness. The bumpiness also affects on the occurrence conditions of the “H-mode” transition.

The control of fueling/recycling is one of important knobs to obtain better plasma performance. A supersonic molecular beam injection (SMBI) is successfully applied to Heliotron J plasma and expands the reachable range in the n_e-W_p space. Moreover, interesting time responses caused by the SMBI are observed, probably suggesting the occurrence of some non-local transport phenomena.

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Gyrokinetic microturbulence investigations towards anomalous optimization

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The GENE/GIST code package [1] was recently developed in order to provide insight to the challenging and largely open issue of the investigation of 3D microturbulence, via coupling of a well-tested geometry module with the (local version of) the nonlinear gyrokinetic code GENE. The code package was successfully benchmarked for both a numerical tokamak [2] and an analytical stellarator equilibrium. GENE/GIST has been applied, in order to characterize ITG microturbulence under the influence of the 3D geometry, to a broad family of stellarator configurations [3] and, further, identify targets for extending present-day stellarator optimizers into including the anomalous transport as well. Along those lines, we examine the impact of two distinct agents on the nonlinear ITG modes, namely: (i) the local shear which restricts the extend of the mode by imposing stabilization and (ii) the zonal flow response, which is apparently connected to the concept of neoclassical optimization [4].

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Investigation of turbulent transport and shear flows in the edge of fusion plasmas

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The understanding of turbulent transport at the plasma edge and on transition to the scrape-off layer is of greatest importance for fusion plasmas. Edge transport sets the values of temperature and density at the pedestal top, parameters which are needed as input for core-turbulence simulations. And SOL transport ultimately defines the peak power density on the divertor plates and the first wall. Furthermore, the interplay between flows and turbulence is one of the most fascinating and rich topics of fusion research.
The talk will give an overview of experimental studies on turbulent transport and the interaction of fluctuations with poloidal flows. The studies were carried out on the low-temperature plasma in the TJ-K stellarator, which is dimensionally similar to fusion edge plasmas. Intermittent structures (blobs) are observed at the edge shear layer and the generation of this structures is discussed. A comparison of the results with measurements obtained at ASDEX Upgrade and HSX confirms the possibility to scale up results from small devices to fusion plasmas.
Results from plasma-biasing experiments give insight into the interplay between fluctuations and $E \times B$ shear flows. Their capability of not only suppressing broad-band turbulence but also enhancing large-scale structures is demonstrated. The observation of increased long-range correlations in the potential with a zonal $m = 0$, $n = 0$ character at low frequencies confirm earlier results from TJ-II [1], which points to a universal feature as to be used for common experimental detection of natural zonal flows.

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Long-range correlations during plasma transitions in the TJ-II stellarator

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Theoretical and experimental work in fusion devices have proved that transport bifurcation to an improved confinement regime is directly related to the formation of sheared flows that can stabilize the turbulence universally present in fusion plasmas as the major cause of plasma transport. The mechanisms governing the development of this bifurcation, which leads to the establishment of a transport barrier, are still one of the main scientific issues facing the magnetic fusion community after more than twenty years of intense research since the discovery of H-mode [1].

Zonal flows have been suggested to explain the Low to High transition (L-H) in magnetic confinement devices [2, 3 and references therein]. Indeed, the existence of zonal flows in toroidal plasmas has been experimentally confirmed [4 and references therein]. In addition, multi-scale physics can be considered a new fingerprint of plasma behaviour during transport bifurcations and as a consequence an essential issue in L-H confinement transitions in fusion plasmas. Thus, the characterization of the emergence of sheared flows and the quantification of the degree of long-range correlation can provide relevant information on the mechanisms involved in the transition to improved confinement regimes.

Recent experimental results related to the emergence of sheared flows in TJ-II have shown the important role of long distance correlation as a first step in the transition to improved confinement regimes as well as the key role of electric fields to amplify them [5]. The new TJ-II experimental set-up with a Li-coated wall [6] and NBI heating have provided evidence of spontaneous bifurcations with the characteristics of transitions to improved confinement regimes (H-mode) [7]. The results presented here reveal the importance of multi-scale physics during transport bifurcations thus providing a critical test for L-H transition models. Dynamic biasing is also being used to externally induce improved confinement regimes and so bring about the long-range correlations observed during such transitions.

On the other hand the damping of sheared flows by atomic physics effects is being investigated by measuring the time decay of the sheared flows under different plasma recycling conditions (i.e. with different neutral densities) in biasing experiments.

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Characteristics of nonlocally-coupled transition of the heat transport in LHD*

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A full understanding of electron and ion heat transport in magnetically-confined toroidal plasmas is crucially necessary to have power over burning fusion plasmas, since the burning plasma is highly autonomous. Nowadays it is common knowledge that the electron and ion heat transport of the magnetically-confined toroidal plasmas is governed by micro-turbulences, which is referred to as “anomalous transport”. Unfortunately, characteristics of the turbulence-driven heat transport are still less well understood due to the existence of incomprehensible phenomena beyond the standard diffusive paradigm. One of the well-known examples of such phenomena is a core electron temperature T_e rise in response to an edge cooling (“nonlocal transport phenomenon”) [1, 2]. On the “nonlocal transport phenomenon”, the nonlocal T_e rise in response to the edge perturbation takes place almost simultaneously at the wider region (e.g. from $\rho = 0$ to $\rho \sim 0.4$) in contrast with the transition to Core Electron Root Confinement (CERC) [3], which usually appears from the center of the plasma and spreads to the adjoining outer region [4]. Recent studies have suggested that a long-ranged fluctuation could have a key role in the nonlocal T_e rise in response to the edge cooling [5]. It should be noted here that a backward transition from the improved state to the low confinement state in the “nonlocal transport phenomenon” still has a spatial uniformity of the change in the electron temperature. It is well-known that the reduction of transport is attributed to the breaking of turbulent eddies and, consequently, the reduction of the radial correlation length (i.e. the disappearance of the nonlocality) [6]. Thus the observation of the nonlocal backward transition suggests that other mechanisms of the reduction of the transport could exist. In the workshop, the characteristics of the nonlocal backward transition in the “nonlocal transport phenomenon”, which would highlight the characteristics of the nonlocally-improved state, will be discussed in more detail.

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Friday

Oct. 16, 2009

08:30 – 09:15	PL04	K. Toi	***Interplay between Energetic Particles and Alfvén Eigenmodes in Toroidal Plasmas***
09:15 – 09:45	I29	D.A. Spong	***Energetic particle-driven instabilities in general toroidal configurations***
09:45 – 10:15	I30	M. Isobe	*Effect of energetic-ion-driven MHD instabilities on energetic-ion-transport …*
10:15 – 10:45	I31	S. Kobayashi	*Energetic particle transport in NBI plasmas of Heliotron J*

Coffee

11:15 – 11:45	I32	S. Murakami	*Optimization study of ICRF heating in the LHD and HSX configurations*
11:45 – 12:15	I33	W.W. Heidbrink	*Measurements of Fast-ion Transport by Instabilities in Tokamaks*
12:15 – 12:45	I34	Y. Suzuki	*3D effect on stochastic field in non-axisymmetric tori*

Closing

Interplay between Energetic Particles and Alfvén Eigenmodes in Toroidal Plasmas

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In toroidal plasmas, various spectral gaps are generated in shear Alfvén continua because of non-uniformity of the magnetic field strength on the magnetic surfaces. Alfvén eigenmodes (AEs) can be easily destabilized by energetic alphas and beam ions in tokamak and stellarator/helical plasmas. These AEs would enhance radial redistribution and/or losse of energetic ions, and lead to noticeable modification of bulk plasma confinement or serious damages of plasma facing components even in low ripple magnetic configurations. Interplay between energetic particles and AEs is an important and interesting physics issue for these toroidal plasmas in reactor relevant conditions. In this talk, the following topics related to AEs in stellarator/helical plasmas are reviewed and discussed, being compared with results from tokamaks: (1) shear Alfvén gap structures and characters of AEs, (2) energetic ion driven AEs in 2D and 3D plasmas with monotonic and non-monotonic rotational transform (or the safety factor) profiles, and (3) enhanced radial transport and/or loss of energetic ions induced by AEs and their effects on bulk plasma confinement.

Energetic particle-driven instabilities in general toroidal configurations

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Energetic particle-driven instabilities have been observed over a wide range of experiments, including tokamaks, stellarators, ST's and RFP's. Interest in this phenomena is motivated by the impacts of energetic particle loss on heating efficiency, concerns about enhanced heat fluxes on plasma facing components, possible diagnostic uses, and the potential of finding methods for more direct channeling of fast ion energy to thermal ions. In order to understand these instabilities, a sequence of models have been developed, including calculation of Alfvén continua [1], mode structures [2], instability drive [3], mode damping rates, and impacts on energetic particle confinement [4]. The additional mode couplings present in three-dimensional configurations such as stellarators and rippled tokmaks increase the number of available Alfvén modes and result in new classes of wave-particle resonance mechanisms. Stellarators encompass a wide range of Alfvén mode structures, ranging from those where poloidal mode couplings dominate (similar to tokamaks), as seen in higher field period, higher aspect ratio devices such as LHD [5] and W7-AS [6], to configurations with equally strong poloidal/toroidal mode couplings (TJ-II, compact stellarators), which induce more strongly 3D Alfvén mode structures. Also, energetic particle-driven instabilities have been observed both with Alfvénic parameter scaling, as well as more recently in HSX [7] and LHD [5] with acoustic wave characteristics. The theory and computational models used to describe these instabilities and their effects will be described and applied to observations from a variety of different experiments.

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Effect of energetic-ion-driven MHD instabilities on energetic-ion-transport in Compact Helical System and Large Helical Device

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Energetic-ion-driven MHD instabilities such as EPM and TAE are of great concern in current fusion experiments because those instabilities may lead to anomalous transport of fast ions/alphas in a future D-T burning plasma. For this topic, impact of energetic-ion-driven MHD modes on energetic-ion transport and/or loss has been experimentally investigated in three-dimensional helical plasmas with NBI. Experiments have been conducted in two helical devices, the small-scale device CHS (R=1 m, a~0.2 m, M/l=8/2) and the large-scale machine LHD (R=3.9 m, a~0.6 m, M/l=10/2); a comprehensive set of fast-particle-diagnostics have been used, i.e. scintillator-type lost-fast-ion probe (SLIP), directional Langmuir probe (DLP) and neutral particle analyzer (NPA). EPMs and TAEs destabilized by co-going transit beam ions were often observed in relatively low density target plasmas [$n_e \approx (0.5\sim2)\times10^{19}$ m^{-3}] of CHS. The fast-particle diagnostics mentioned above revealed that beam ions are anomalously transported toward the outboard side of the torus due to those bursting MHD modes. Beam-ion-loss rate to SLIP and DLP increases as the magnetic fluctuation amplitude increases. Guiding center orbit simulation considering possible level (dB~10^{5}-10^{4} T) of magnetic fluctuation suggests that orbits of co-going beam ions tend to be expanded toward the outboard side of the torus due to the presence of magnetic fluctuation and go across the last closed flux surface consequently. Also, in LHD, non-classical transport of co-going beam ions due to bursting TAEs has been so far recognized [1]. To enhance understanding of beam-ion behavior while fast ion driven instabilities occur in l=2 helical system, SLIP was lately installed at the outboard side of LHD [2]. The SLIP indicates that correlated with repetitive bursting fluctuation, co-going beam ions having energy of ~150 keV are expelled toward the outboard side. Energy of detected fast ions is consistent with energy of anomalously transported beam ions measured with NPA. In this paper, the impact of EPM and TAE on fast-ion transport in CHS and LHD will be reported.

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Energetic particle transport in NBI plasmas of Heliotron J

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Energetic particle confinement and control of the neoclassical transport in 1/ν regime are important issues in helical/stellarator magnetic configurations because burning plasma is mainly heated by the energetic particles produced by fusion reactions. Study of the energetic-particle-driven MHD instability is also an important issue since it may affect the transport of energetic particle as an anomalous loss.

The configuration effect on the energetic particle confinement has been studied in Heliotron J with regard to bumpy magnetic field component, being the toroidal mirror ratio which is the key factor of the drift optimization of the Heliotron J magnetic configuration. The configuration scan experiments have been carried out by changing the bumpy magnetic field with keeping plasma volume, plasma axis position and edge rotational transform almost constant [1]. It has been found that the 1/e decay time of high energy CX flux after the NB turned-off has increased with bumpiness. The Fokker-Planck analysis shows the effective loss time in the high bumpiness configuration has been longer than that for the medium and low bumpiness cases [2]. These results indicate that the effective confinement of the energetic ions improves as bumpiness increases.

Global Alfvén eighenmodes (GAE) have been a candidate of most unstable modes in the low-shear-magnetic-configuration of Heliotron J when fast ion pressure has became fairly high. Recently a directional probe method has been applied to Heliotron J plasma in order to investigate the fast ion behavior during GAE [3]. The co-directed ion flux synchronized with GAE bursts has been observed in the co-directed NBI plasmas and it has been sensitive to the burst interval and amplitude. On the contrary, the response of the counter-going ion flux to GAE burst was weak. Strong bursting GAE has not been observed in the low bumpiness configuration. The interaction between the fast ion loss and the MHD activities in the Heliotron J NBI plasmas will be discussed.

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Optimization study of ICRF heating in the LHD and HSX configurations

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ICRF heating experiments has been successfully done in helical systems and have demonstrated the effectiveness of this heating method in three-dimensional (3D) magnetic configurations. In LHD, significant performances of this method have also been demonstrated and up to 2.5MeV of energetic tail ions have been observed by fast neutral particle analysis (NPA). These measured results indicate a good property of energetic ion confinement in helical systems. However, the measured information by NPA is obtained as an integrated value along a line of sight and we need a reliable theoretical model for reproducing the energetic ion distribution to discuss the confinement of energetic ions accurately.

To solve this problem we have studied ICRF heating in the LHD combining two simulation codes: a full wave solver TASK/WM[1] and a 5-D drift kinetic equation solver GNET[2]. Characteristics of energetic ion distributions in the phase space are investigated changing the resonance heating position; i.e. the on-axis and off-axis heating cases. The simulation results are also compared with experimental results in the two heating cases evaluating the count number of the neutral particle analyzer and a relatively good agreement is obtained.

On the other hand, recent numerical studies of energetic ion confinements in the LHD configurations indicate that an optimized configuration of the energetic ion confinement is different from that of the neoclassical transport (R_{ax}=3.53m)[3] due to the finite orbit effect of the energetic ions. The similar tendency is also observed in the HSX configurations. Additionally the previous simulation study of ICRF heating in LHD[2] shows that the stable trapped particles near the resonance surface play an important role in generating energetic tail ions. Thus there is a possibility to find a better configuration and heating scenario than the present ones, and it is interest to investigate an optimization of ICRF heating in point of views of the energetic tail ion generation and their confinements in a helical plasma.

In this paper optimizations of the configurations and the heating scenario of ICRF heating are investigated in the LHD and HSX plasmas applying two global simulation codes; a full wave solver TASK/WM and a 5-D drift kinetic equation solver GNET. The difference between the ICRF heating optimization and the neoclassical transport one is discussed.

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Measurements of Fast-ion Transport by Instabilities in Tokamaks

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Several topics of relevance to stellarator research are highlighted. First, above a certain threshold, the helical field perturbations associated with tearing modes cause large fast ion transport [1]. A model [2] that attributes the losses to island overlap in drift-orbit phase space agrees well with the measurements. Next, a new type of charge-exchange recombination spectroscopy called fast-ion D-alpha (FIDA) [3] provides velocity and radial information about the fast-ion distribution function. In low temperature MHD-quiescent plasmas, the FIDA measurements agree well with classical predictions [4] but discrepancies are observed in plasmas with Alfvén instabilities or large plasma temperature. In plasmas with many low-amplitude ($\delta B/B \sim 10^{-4}$) Alfvén modes, the central fast-ion profile is flattened by the instabilities [5]. In plasmas where the ratio of fast-ion energy E to plasma temperature T is relatively low ($E/T \lesssim 10$), fast-ion transport attributed to microturbulence is observed [6].

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3D effect on stochastic field in non-axisymmetric tori

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The stochastization due to the 3D effect is an intrinsic property in stellarators. The 3D effect is a reaction of the pressure-induced perturbed field, which is produced by the parallel current flow along 3D field lines. Since the stochastization leads the degradation of the confinement, the study of the stochastization is important to aim the reactor.

Most important parallel current in stellarators is the Pfrish-Schlüter (P-S) current to keep the equilibrium force balance. Using 3D MHD equilibrium codes without the assumption of nested flux surfaces in *a priori*, which are HINT/HINT2 and PIES codes, the 3D effect was studies. In a conventional heliotron, field lines in the peripheral region became strongly stochastic for the net-current free equilibrium. In an advanced stellarator, the stochastization was not large but large magnetic island appeared because of the weak magnetic shear. Sometimes, the 3D effect suppressed the and magnetic island, so-called 'self-healing'. In previous studies of the 3D effect, the parallel current was assumed only the P-S current. However, since the net-toroidal current was observed in many experiments, the study of the 3D effects driven by other parallel currents, which are bootstrap current, NBCD and ECCD, is an urgent issue.

On the other hand, the effect of the stochastic field is also a hot topic in tokamak studies. Superposing the stochastic field to the plasma, there is a possibility to mitigate or eliminate the edge localized mode (ELM). To model the field configuration superposed the stochastic field, the vacuum approximation, which is the 2D MHD equilibrium with nested flux surface superposed the vacuum 3D perturbed field, are widely used. However, this model does not include the response from the plasma and its stochasticity is always fixed for the vacuum. Since it is expected the parallel current flow along rippled field lines causes the 3D effects, the study of the 3D effect is also an important issue in tokamaks as well as stellarators.

In this study, we discuss the stochastization due to the 3D effect in stellarators and tokamaks. To model the 3D effect and consider the stochastization qualitatively, 3D MHD equilibrium is studied theoretically in stellarators and tokamaks. Special notice is the 3D effect due to the net toroidal current in stellarators and tokamaks.