stellarator news

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Around the Labs

Configuration experiments in ATF

The ATF experiment has concentrated on low-collisionality transport in ECH plasmas. Trapped electron modes should be stabilized by shear and by reduction of the confined trapped particle population, while interchange modes are stabilized by the magnetic well. In ATF, these physical quantities can be varied by using external poloidal coils to change the axisymmetric dipole and quadrupole magnetic field components, which changes the magnetic axis position, flux surface shape, rotational transform (ɫ), and helical field ripple geometry. The magnetic configuration can be varied from shot to shot or dynamically during a single discharge. These techniques have been exploited to verify the neoclassical theory of the bootstrap current. Recently the dynamic configuration control capability has been expanded significantly.

Figure shows a contour map of the configuration space in ATF. Three parameters are used to characterized the configuration space: (1) fraction of trapped particles confined (f_{TPC}), which is related to the closing of the constant-$|B|_{min}$ contours in the plasma region[1]; (2) magnetic shear $\hat{s} = -(\rho/ɫ)(dɫ/d\rho)$ at a normalized average minor radius ρ between 0.4 and 0.6; and (3) magnetic well radius $\rho(V''= 0)$ for volum- average beta of 0.1% (typical for the ECH plasmas). Although contours of constant $\hat{s}$ and $\rho(V''= 0)$ nearly coincide for the low-beta plasmas, contours of constant f_{TPC} confined are nearly orthogonal to the others over much of the range shown. This means that we may be able to separate the effects of the trapped particle population from the effects of magnetic shear or well/hill. The overlaid lines in the figure are the configuration scans carried out in recent ATF experiments. Preliminary analysis of the f_{TPC} scan (with constant shear and well

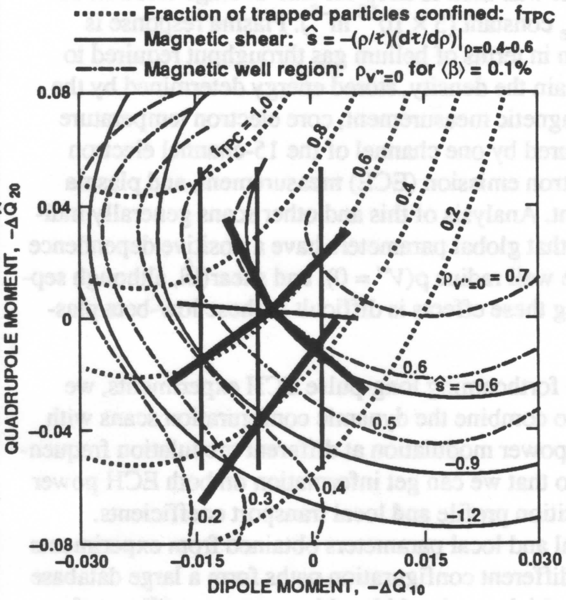

Fig. 1. The ATF configuration space.

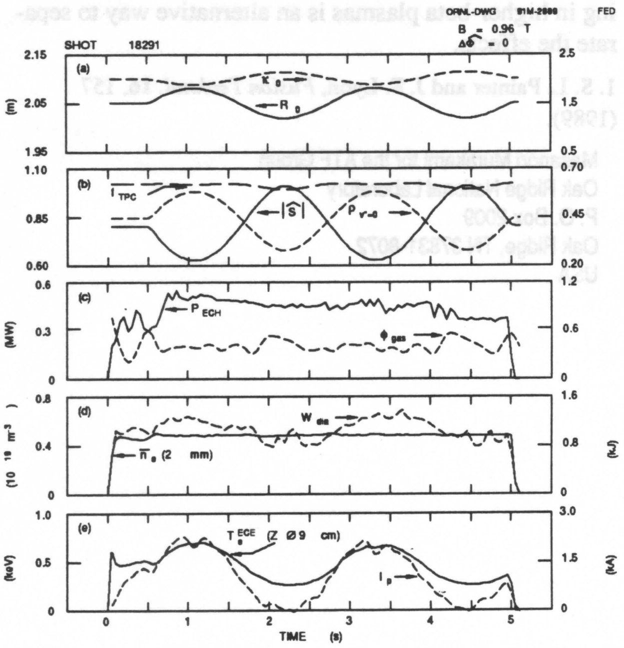

Fig. 2. Configuration modulation experiments in ATF.

tail component, which is presumably from the minority hydrogen. This bulk temperature is roughly consistent with the diamagnetic measurements. Thomson scattering measurements for the electron temperature profile are scheduled in May.

Because the present antenna position prevents experiments with a wide range of magnetic axis shift, the first phase of the ICRF heating experiment is scheduled to last until the summer, when the antennas will be removed. The results of the first phase experiments will be examined to determine the schedule of second phase experiments with higher power heating, which will possibly take place in 1992.

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Edge plasma studies in Heliotron-E

The main objectives of the Heliotron-E edge plasma studies are to understand the complicated three-dimensional plasma edge and to improve the core plasma performance through actively controlling the edge region.

Since the "divertor" flux bundles are terminated at the metal chamber wall, it is not necessary to keep the equality between the electron and the ion currents to the surface in a local small area. The edge plasma transport might be affected by the electric field due to this non-ambipolar flow in the edge region. In order to study this effect, the ion saturation current and the floating potential to the wall have been measured by using new poloidal calorimeter arrays of electrically insulated plates located at four different cross-sections in one helical pitch of the torus.

The localization of the ion saturation current and the wall heat load on the divertor trace was confirmed for different four toroidal positions and non-uniformity of these values along the toroidal direction was also observed. The absolute value and also the polarity of the floating potential varies in the poloidal and toroidal directions. This suggests that the charge equality of the plasma flow is broken in a local area. Since it was observed that the polarity of the floating potential of many (but not all) plates was consistent with the direction of ∇B drift and was reversed by the change of the helical field direction, it is considered that ∇B drift has an important part in this flow. In order to actively control the edge plasma potential, we are planning a biased limiter experiment with two pairs of material limiters, on the low and high field sides. Related to this limiter experiment, using a new material is also proposed to control recycling on the limiter surface.

Density clamping is usually observed in Heliotron-E ECH-plasmas and this prevents control of the plasma density. This phenomenon is characterized by a burst of edge plasma, i.e. an abrupt increase in the intensity of ion saturation current (and also the density) at the edge of the core plasma and in the divertor trace, together with increased H_{alpha} emission during the RF pulse. At almost the same time, the line-averaged density starts to decrease. Once this has happened, it is hard to recover or increase the density by gas puffing.

The manner of change in the floating potential of the calorimeter plate when density clamping occurs is not the same for each plate. The most interesting observation is a spike-like change in the potential at some positions. Both negative and positive spikes were observed, depending on the positions, but the largest one is negative. On the other hand, the potential at the edge of the core plasma ($r - r_s = 0.01$ m) showed a positive spike. The electron temperature estimated with the triple probe method increased at this time, but after that, the temperature decreased to the previous level or lower. These observations suggest that there are two confinement modes in the ECH phase. And it is supposed that the transition from one mode to another causes the spike-like change in edge potential and temperature.

The uncontrollable density decay itself can be reduced by superimposing a toroidal field. However, the burst of edge plasma and increase of H_{alpha} emission are still observed here, i.e. the loss rate is not reduced. In this case, the outer plasma edge might be in contact with bumper limiters or other materials on the high field side. This suggests that it is not the gas puffing at the low field side, but instead the increase of gas feed due to change of recycling or fueling efficiency caused by contact with materials at the high field side, that improves the situation.

Recently, a new clamping-free mode was observed even in the standard configuration. In this mode, not only was the density kept high, but also the rapid increase of H_{alpha} emission and the ion saturation current was not observed. The large negative spike in the floating potential also disappeared. Since the electron temperature of the core plasma was almost the same as that in the low density case with clamping, the internal energy of the confined plasma increased in this mode.

radius) and other scans indicates that global parameters are insensitive to f_{TPC} (i.e., "orbit optimization") until its value becomes > 0.9. Figure 2 shows an example of configuration modulation experiments in a 5-s ECH discharge. Both shear and magnetic well radius are modulated sinusoidally, while f_{TPC} remains constant. Density feedback control was used to keep the line-average electron density $\bar{n}_e$ constant (5×10^{18} m^{-3}). Plasma response is shown in terms of helium gas throughput required to maintain the density, stored energy determined by the diamagnetic measurement, core electron temperature measured by one channel of the 15-channel electron cyclotron emission (ECE) measurement, and plasma current. Analysis of this and other scans generally indicates that global parameters have a positive dependence on the well radius $\rho(V'' = 0)$, and shear($|\hat{s}|$, although separating these effects is difficult in these low-beta plasmas.

In the forthcoming long-pulse ECH experiments, we plan to combine the dynamic configuration scans with ECH power modulation at different modulation frequencies so that we can get information on both ECH power deposition profile and local transport coefficients. Global and local parameters obtained from experiments with different configuration paths form a large database from which we should be able to separate effects of magnetic shear and curvature. Magnetic well broadening in higher-beta plasmas is an alternative way to separate the effects.

1. S. L. Painter and J. F. Lyon, *Fusion Technol.* **16**, 157 (1989).

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ICRF Heating Experiments in CHS

From the beginning of CHS experiment, the Nagoya Type-III antenna has been installed for the purpose of ion Bernstein wave heating and has been used mainly for the initial production of a target plasma for NBI heating experiments. A heating effect was also observed with this antenna but strong heating has not yet been achieved.

Two new half-turn antennas were installed in March to investigate the availabilities of ICRF heating (fast wave) for the low-aspect-ratio helical system. The antenna conductor lies on the poloidal cross section and is located at the high field side of torsatron field (beneath the helical conductor). The toroidal position of the antenna is at 13.75 degrees away from the poloidal cross section where the magnetic surfaces are horizontally elongated (one helical period covers 45 degrees). The two antennas are identical and installed at the same toroidal positions in adjacent helical periods.

The antennas are designed to fit the outermost magnetic surface of the configuration with an inward shifted magnetic axis (R_{ax} = 92.1 cm). Although ordinarily there is little room beneath the helical conductor, an inward shifted configuration makes room for the antennas at the outboard side of vacuum chamber. It is also expected that inward shifted configuration gives better coincidence of the magnetic surfaces and the drift orbits of helically trapped particles.

Two set of transmitters were used to supply independent rf power (20 ms) to each antenna. The frequency was fixed at 14 MHz throughout the experiments and the magnetic field strength was varied to determine the optimum resonance condition. Deuterium gas with 10% hydrogen was used. ICRF was applied to the plasma which was first produced by 28-GHz ECH at about 0.9 T magnetic field. Significant heating effects are observed when the ion cyclotron fundamental resonance layer for protons at 14 MHz lies within the region of about one third of plasma radius. Simultaneous ECH heating is not necessary for sustaining the plasma. The loading of ICRF power is about 20% higher when the ECH is turned off during ICRF heating. The maximum diamagnetic energy was 650 J for a plasma density of 2×10^{13} cm^{-3} with a net rf loading of 300 kW total. The relative phases of the antenna currents are controlled. The apparent heating efficiency varied about 30%.

The two-component ion energy spectrum was obtained with the neutral particle energy analyzer. The bulk component shows a 350-eV distribution with about a 1-keV

Up to now, this mode is out of our control but it seems that the key parameter is a small change of the filling pressure (or the recycling condition) just before RF pulse or the pre-ionization condition relating to the existence of non-thermal electrons.

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Ion heat conductivity, radial electric fields, and CX losses in the W7-AS stellarator

The energy balance analysis for discharges heated with both ECR (up to 1 MW at 70 GHz) and NBI (up to 1.6 MW with tangential injection) in the W7-AS stellarator ($B = 2.5$ T, $R = 2$ m, $a < 18$ cm, 5 field periods) is based on measured profiles: electron density and temperature by Thomson scattering, ion temperature by active charge exchange (CX) neutral particle analysis and by CX recombination spectroscopy (CXRS). The experimental estimate of the local T_i is higher than the real value for higher densities $n_e > 3 \times 10^{13}$ cm^{-3} owing to the attenuation of the outgoing neutral flux. A standard procedure is used for correction of the T_i profile. For the ECRH discharges with lower density the correction is small.

The electron and ion energy balance are analyzed independently. This procedure becomes critical at higher densities where the collisional coupling of electrons and ions, determined by the absolute difference of $T_e(r)$ and $T_i(r)$, dominates the energy balance. At higher densities $n_e > 10^{14}$ cm^{-3}, where the electron and ion temperatures are nearly identical, the total energy balance can be considered, since the electron-ion coupling disappears.

The ECR power deposition, which is highly localized close to the resonance zone, is modeled analytically corresponding to 3-D ray tracing calculations. The NBI electron and ion heating profiles are calculated by the FAFNER 3-D Monte Carlo code. For tangential injection in W7-AS with high beam absorption, fast-ion orbit losses are very small. Also, the reduction of NBI heating efficiency due to fast CX losses is negligible in the bulk part of the plasma. This is consistent with DEGAS simulations[1] of the neutral gas profiles. For NBI discharges at higher densities, the CX losses turn out to be rather small compared to the ion transport losses in the bulk part of the plasma. In ECRH discharges at lower densities, however, where the collisional ion heating is rather small, the ion energy balance is significantly affected by the CX losses.

The radial electric fields calculated by the condition of ambipolarity of the neoclassical particle fluxes (with only one ion species included) turned out to be rather small. This is consistent with first observations of the Doppler shift of impurity lines (CXRS). As in cases with the ion root ($E_r < 0$), the ambipolar fluxes are mainly determined by the electrons, and the discrepancy between experimental and neoclassical electron heat conduction plays a minor role for the reliability of the radial electric field estimation. More sensitive for the E_r estimation is the use of only one ion species in the ambipolarity condition. Including the radial electric field in the neoclassical models results in a significant reduction of the predicted ion heat condution.

The experimental values obtained from the ion energy balance are typically between these two theoretical cases. Within the errors in estimating the electron-ion power transfer for moderate and higher densities, the electron heat transport is independent of the heating method (ECRH, NBI, and combined). Within the reliability of these estimates, the electron heat transport plays only a minor role in the NBI-heated discharges. Together with strong radiative power losses the ion energy transport is dominant; this is roughly of the order of the neoclassical prediction.

The ion transport on W7-AS has not yet reached the low level obtained in the W7-A stellarator where much higher ion temperatures (up to 1 keV) were achieved. In W7-A the nearly perpendicular NBI significantly affected the particle balance, leading to very strong radial electric fields[2] which essentially improved the thermal ion energy confinement. Further investigations in W7-AS with tangential NBI will concentrate on the role of the radial electric field on the confinement properties.

1. F. Sardei et al., this Stellarator News

2. H. Wobig et al., in *Plasma Physics and Controlled Nuclear Fusion Researsc (proc. 11th Int. Conf. Kyoto, (1986)*, Vol. 2, IAEA, Vienna (1987) 369.

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Particle transport and plasma edge behavior in the W7-AS stellarator

The particle transport in W7-AS is investigated by coupling DEGAS code simulations with H_{alpha} emissions measured at different toroidal positions. Radially resolved ion fluxes are obtained from the calculated ion source distributions, after calibrating the recycling fluxes with the H_{alpha} signals. Additionally, for NBI discharges, radial profiles of the neutral beam halo and of the beam deposition are taken from FAFNER code calculations.

Several NBI discharges at 2.5 T, with 350 and 700 kW of heating power and different plasma densities and limiter apertures, have been analyzed and compared with ECRH discharges. Flux-surface-averaged neutral densities drop from typically 10^{10} cm^{-3} at the limiter to 10^{7}–10^{9} cm^{-3} at the plasma center, depending on the plasma density and limiter position. For these neutral density profiles, charge exchange (CX) with thermal ions represents a substantial ion power loss channel for low-density ECRH discharges, whereas CX with the fast NBI ions is negligible. On the other hand, CX with impurities may result in an enhancement of the total radiation power losses by more than a factor of two, if low-Z impurities are dominant.

For NBI discharges, the central plasma refueling is dominated by beam deposition, but the global plasma density level of both NBI and ECRH discharges is sustained by recycling processes, which provide more than 90% of the particle fluxes in the plasma density gradient region and at the plasma edge.

A comparison of the ion fluxes, as simulated by the DEGAS code, with neoclassical predictions including a self-consistent ambipolar radial electric field shows a reasonable agreement in the bulk of the plasma but a large discrepancy at the plasma edge. The ECRH and NBI discharges analyzed indicate that the gap between the two fluxes increases with decreasing density.

Radial diffusion coefficients at the limiter, as derived from DEGAS code calculations, were compared to Langmuir probe data for ECRH- and NBI-heated discharges with $\iota_a \approx 0.34$ and minimum or maximum limiter apertures. From a scan of the edge density between 10^{12} and 10^{14} cm^{-3}, with electron temperatures in the range of 40 to 120 eV, both the probe measurements and the H_{alpha} calibrated simulations suggest a scaling $D_a \sim n_{ea}^{-1/2}$, which seems to be independent of the heating method and limiter aperture. The absolute values of D_a are below the Bohm values by factors of 2–5.

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Design Studies

Optimization of coils for divertor experimentation in W7-X

The boundary region on the outer side of the torus of the Helias W7-X configuration (D23P4, R = 5.5 m, aspect ratio A = 10, five field periods, rotational transform $0.83 < \iota < 1$) appears to be suitable for divertor action. There the plasma boundary forms half-helix-like edges, close to which X-points between islands appear. Detailed studies[1] of the vacuum field outside the last closed surface show that the field line diversion occurs essentially along the half-helix-like edges, so that the divertor equipment has to be installed in these areas, with the consequence that a sufficiently large distance between the plasma and the cryostat containing the superconducting modular coil structure is required there.

The modular coils shown in Fig. 1 have been obtained by using the NESCOIL code and optimizing, observing the following geometrical constraints:

- minimum and maximum plasma-filament distances of 0.4 and 1.0 m, respectively,
- minimum filament-filament distance d = 0.26 m, and
- minimum curvature radius of filaments r_{curv} = 0.44 m,
- a radial extent of ~0.1 m for the divertor structure,
- a plasma-divertor distance of ~0.1 m, and
- a spacing of ~0.18 m required for the cryostat.

The number of coils per period (= 8) is determined by the admissible magnetic ripple, e.g., by a value of less than 10^{-3} on the magnetic axis, in order to ensure that the transport properties are not affected by this field variation. The ripple decreases exponentially with decreasing coil aspect ratio A_c so that a large plasma-

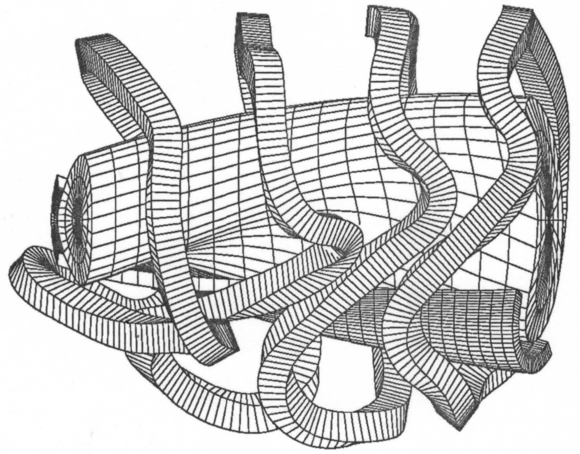

Fig. 1. Half a period of an optimized coil configuration with a divertor trough along the half-helix-like edge. The number of coils per period $N = 8$, and the coil cross section $q = 0.22 \times 0.22$ m^2. The current density $j = 48$ MA/m^2, and the magnetic field $B = 3$ T.

coil spacing allows the number of coils per period to be reduced.

1. E. Strumberger, in this issue of *Stellarator News.*

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Structure of the magnetic field line diversion in Helias configurations

The vacuum magnetic field outside the last closed magnetic surface of Helias configurations similar to that shown in Ref. 1 is investigated with respect to its field line diversion properties. In a Helias configuration with *n* periods, *n* half-helix-like edges run along the toroidally outward side of the plasma boundary and yield the possibility of separatrix formation owing to the coincidence of the helical edge and X-points between islands. With the choice $n = 5$ and $\iota = 1$ at the plasma boundary, there are five magnetic islands outside the last closed magnetic surface. With a homogeneous distribution of points on a surface outside but very close to the plasma boundary as starting points, magnetic field lines are traced until they cross the surface of the divertor trough. It is positioned at an average distance of approximately 20% of the plasma radius from the plasma boundary, and its shape is chosen to be approximately parallel to the field lines. Its poloidal extent leads to complete deposition of all field lines (there are no field lines crossing the first wall).

In a very simplified scrape-off layer (SOL) simulation, field lines are subjected to random displacements after characteristic mean free paths. These displacements simulate an anomalous cross-field transport coefficient D of order 1 m^2/s, acting on particles with a characteristic velocity along field lines of the order of the sound speed. Under these conditions, persistence of the diversion process at the helical edge, a marked blurring of detailed island structures, and, consequently, an increase in plasma-control surface interaction area by more than an order of magnitude as compared to results without displacements are found. For $0.2 < D < 1$ m^2/s, intersection areas of 3–14% of the plasma surface are obtained. The average field line length L until intersection decreases with increasing diffusion to about two toroidal transits, while the intersection angles remain at about 4°.

1. P. Merkel, in this issue of *Stellarator News.*

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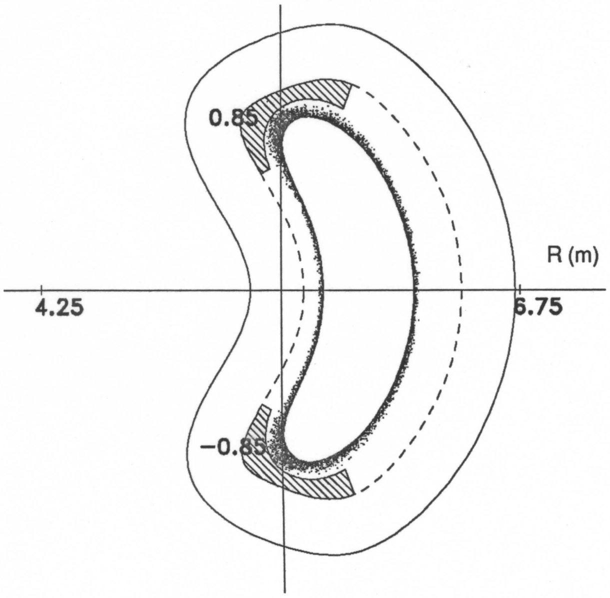

SOL simulation with a field line diffusion model; the shaded areas show cross sections of the half-helix-like divertor troughs.

On natural islands and the edge structure of the W7-X stellarator

Natural magnetic islands and the edge structure of nine configurations of W7-X vacuum fields, derived by parameter variation of a consistent data set,[1] are investigated by field line tracing as well as by a Monte Carlo guiding center code. In each configuration only one set of main islands exists, with ι_i = 5/6, 5/5, or 5/4 and moderate size; higher-order islands are much smaller. One can control the size and phase of the islands by small stationary currents in correction coils; effects of symmetry-breaking perturbations can be substantially reduced by toroidally matched currents. Analysis of fixed points yields valuable quantitative information[2] for defining a divertor for W7-X, which must be compatible with requirements of access to the experiment and permissible heat loads.

Regarding the first wall of the vacuum vessel (chosen at a constant radial distance $\Delta = 18$ cm inside the coil bores) as a boundary, in the system of nine configurations, W7X-A to W7X-I, there are four separatrix-dominated cases, providing a divertor topology *per se*. The use of external islands, preferentially with $\iota = 5/4$, is an interesting option in addition to the presence of an ergodic edge layer. The other five cases are originally limited by the wall. Separatrix ergodization by plasma currents, finite beta, and/or external perturbations can remove the closed surfaces outside the main islands and thus introduce a divertor. If required, additional perturbations can be provided by the correction coils. In the unperturbed vacuum fields studied so far, "hot spots" are obtained at the first wall of the vessel, basically similar to the helical stripes found in the W7-AS stellarator. For the torus of W7-X, or for a system of simple divertor target plates, the total intersection area is estimated at $\lesssim 1$ m^2, resulting in a considerable heat load for a total power of the order of 10 MW. This heat load can be reduced by small oscillating currents in the correction coils; further reduction is envisaged by improved shaping of the target plates.

1. J. Kisslinger et al., "Magnetic Field and Coil Systems of the Modular Helias Configuration HS~5-10," in *Fusion Technology: Proceedings of the 16th Symposium, London, 1990*, in press.

2. F. Herrnegger, in this issue of *Stellarator News*.

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Angle between the hyperbola-asymptotes along a closed divertor field line

Closed magnetic field lines (fixed lines) are associated with rational values m/n of the rotational transform ι, e.g., $\iota = m/n = 5/5$. These fixed lines close upon themselves for an $m = 5$ periodic stellarator configuration after one toroidal and one poloidal revolution. If the residue R* of that fixed line is negative and in the range $\{-1, 0\}$, then the neighborhood of the fixed line is hyperbolic (X-line), where the angle γ between the asymptotes of the hyperbolas (as seen in planes φ = constant from the interior of the nested magnetic surfaces) varies along the fixed lines. The knowledge of γ is important for the understanding of the X-divertor in a stellarator[1]. The figure shows the angle γ as function of the toroidal angle φ for the $m/n = 5/5$ fixed line in the reference configuration W7X-A. The field line starts in the bean-shaped cross section at the inboard side. The main minima of γ appear at the top and bottom of the bean cross section and on the outboard side of the triangular cross section.

1. F. Rau, in this issue of *Stellarator News*.

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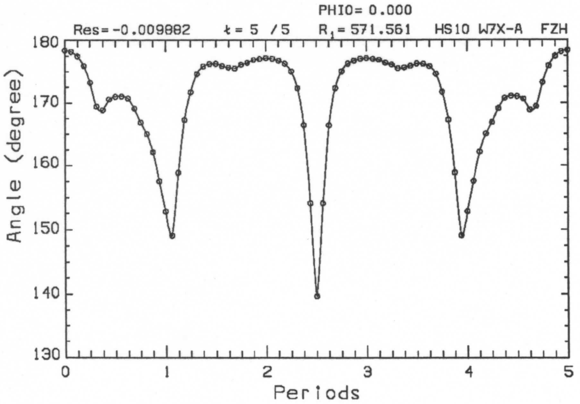

Angle of asymptotes in Wendelstein 7X-A as a function of the toroidal angle.

On the existence and uniqueness of dissipative plasma equilibria in a toroidal domain

The question of existence and uniqueness of plasma equilibria in a toroidal domain is still in part open and of great theoretical interest. Moreover, a deeper theoretical understanding of this problem might help to clarify certain aspects of the experimental results.

As highlighted by Grad, ideal, nonpathologic magnetohydrodynamic (MHD) equilibria are unlikely to exist in the absence of axial symmetry. On the other hand, one can conjecture, on the basis of the extensive literature concerning the theory of dissipative flow, that the mathematical pathologies can be due to the ideal character of the model which Grad considered, and that the lack of symmetry can affect, of course, the shape of the solution, but does not prevent a solution from existing.

Bearing this conjecture in mind, we have considered a one-fluid, dissipative model of MHD plasma equilibrium. The equations include inertial forces, finite resistivity and viscosity, and a particle source that sustains the pressure gradient in the plasma; viscosity is described by the Braginskii operator. Resistivity, viscosity and plasma density are assumed to be uniform. This dissipative model eliminates the topological constraints of the ideal MHD model (line integral dl/B = const on rational magnetic surfaces) and is also applicable to non-axisymmetric equilibria with magnetic islands and stochastic regions. In particular, the model applies to the boundary regions of stellarators and tokamaks, where the plasma is collision dominated and magnetic surfaces are destroyed.

A boundary-value problem in a general 3-D toroidal domain is formulated, and the mathematical treatment follows the framework of functional analysis which has been successfully used in hydrodynamics to prove the existence of stationary solutions of the Navier-Stokes equations. The main results obtained so far are:

We prove the existence of at least one weak solution if the plasma source (or beta) is sufficiently small, or resistivity and viscosity are sufficiently large;

we obtain an estimate (in norm) of the solution;

we prove that, under a condition for existence similar to that mentioned above, but more stringent, there exists only one solution;

the condition for uniqueness holding for hydrodynamic stationary flow is obtained when the magnetic field is dropped.

If resistivity or viscosity becomes small, existence and uniqueness of the solution are no longer guaranteed, and bifurcated equilibria may occur.

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People

ATF graduate students win MFE post-docs

Two ORNL graduate students from the Nuclear Engineering Program at the Georgia Institute of Technology have been awarded United States Department of Energy Magnetic Fusion Energy Post-Doctoral Fellowships. The prestigious MFE fellowship awards are for two years.

Mr. Greg Hanson and Mr. Mickey Wade are pursuing their doctoral degrees performing experimental work on the ATF stellarator at ORNL. Both students are expected to receive their doctoral degrees in Nuclear Engineering (Fusion Option) from Georgia Tech on September 6, 1991.

Mr. Wade has been working on his thesis at ORNL since January 1989. His thesis research on ATF has involved the use of a neutral particle analyzer (NPA) to analyze and understand the confinement of both fast and thermal ions on the ATF stellarator. Mr. Wade has worked very closely with Dr. Dick Colchin on this work, which has resulted in fundamental insight into ATF ion confinement.

Mr. Hanson has been working on ATF since March 1989. His thesis research involves the use of reflectometry to measure density fluctuations in the outer region of the confined plasma in ATF. This work is in an effort to increase understanding of the role of turbulent transport in stellarators. Mr. Hanson has measured both fluctuation levels and correlation lengths of the turbulence in ATF over the outer portion of the ATF plasma. He has worked closely with Dr. Jeff Harris, Dr. John Wilgen, Dr. Nicholas Dominguez, and Dr. Ben Carreras to develop the reflectometer and quantify and identify the turbulent modes in the ATF plasma. The reflectometer design is unique in its operation as a quadrature interferometer, and the resulting work has considerably enhanced the understanding of instability modes under varying conditions in the ATF plasma.