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U.S. Stellarator Community response to FEAC Panel

In January 1996, the Fusion Energy Advisory Committee (FEAC) recommended to DOE that a review of alternative concepts (including stellarators) be carried out as part of making the transition to a Fusion Energy Sciences Program. This review was held on April 23, and the U.S. stellarator community presented its response to the questions raised by the review panel. The response includes a review of the worldwide status of research and achievements on stellarators, an evaluation of the appropriate level of stellarator research in the United States, and the potential impact of U.S. stellarator research on fusion as a whole.

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Real-time boronization experiments in CHS

Real-time boronization (RTB) by puffing decaborane into the main neutral beam injection (NBI)-heated plasma is a promising wall-conditioning technique for use in the Large Helical Device (LHD), which employs steady-state high magnetic fields produced by superconducting magnets. The Compact Helical System (CHS) has been used as a test bed to study RTB [1]. Decaborane ($B_{10}H_{14}$) is a

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less hazardous carbon-free boride with a low H/B atomic ratio. Gas puffing RTB produces excellent results because of its ability to control the amount of decaborane used, and its localized gas injection. Because RTB actively utilizes the high magnetic field and the high heating power for the main plasma, the fueling efficiency and film quality resulting from boronization in a machine such as LHD are expected to be better that those achieved with conventional methods such as glow discharges.

The results of the RTB experiments on CHS are summarized as follows:

- The amount of decaborane required to reduce plasma impurities, such as oxygen and metals, was two orders of magnitude less than that required using conventional techniques, resulting in expansion of the operative region of the plasma density and the stored energy.
- Injection just inside the last closed magnetic flux surface (LCFS) was efficient in CHS. However, injection just outside of the LCFS is recommended as the optimum position in LHD to avoid high heat flux on the puffing nozzle.
- Even after RTB with the wall at room temperature, hydrogen recycling did not increase, probably owing to the small amount of decaborane used and the high heating power of the main plasma.

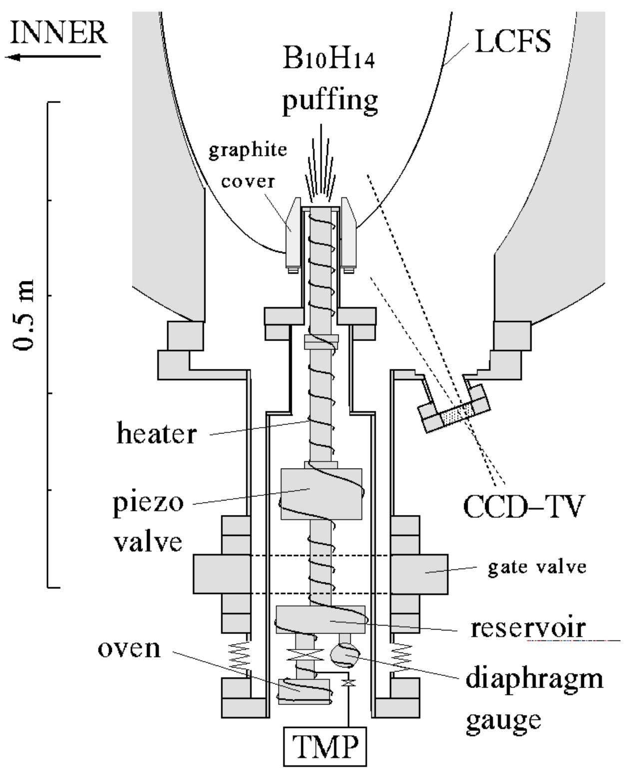

Fig. 1. The movable ga -puffing RTB device in CHS.

As shown in Fig. 1, the movable gas puffing device was installed at the bottom of the CHS vacuum vessel [2]. The puffer system was operated at 120°C to obtain a vapor pressure of decaborane as high as 40 torr. The injected gas amount per shot, which was typically 5 torr-L/s for 50 ms, was measured from the pressure change at the reservoir. For this work, CHS was operated at a fixed magnetic field strength B_T = 1.2 T with the plasma major radius R_{ax} = 0.921 m, where the LCFS touches the inner side wall of the stainless steel chamber as a kind of limiter. The port-through NBI heating power was 0.8 MW. By changing the decaborane puffing position, two types of experiments were performed: injections from 20 mm inside the LCFS (7% of the plasma minor radius, a_p) and 125 mm outside the LCFS (45% of a_p), while the main H_2 gas puffing position was fixed at the chamber wall.

Reduction of plasma impurities

Figure 2 clearly shows that, during RTB with injection inside the LCFS, the O V intensity decreased shot by shot and, in about 10 shots, attained the level achieved after RTB. Here the wall condition “before RTB” means the condition before titanium flashing, which is always required a few times each day to overcome radiation collapse at high plasma density which can be seen in Fig. 2 [3]. RTB suppressed the oxygen impurity, resulting in expansion of the operative region of the plasma density and the stored energy towards levels predicted in CHS by

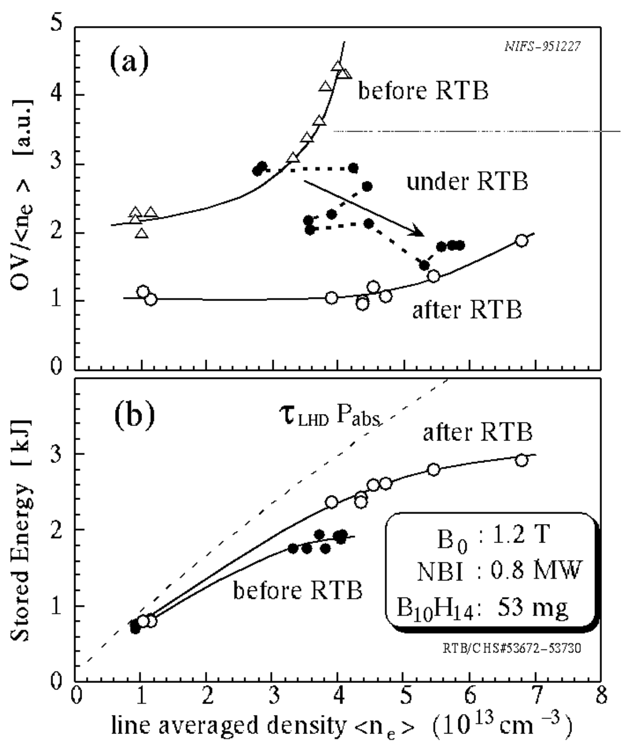

Fig 2. (a) The O V intensity and (b) the plasma stored energy (the broken line represents the quantity calculated with LHD scaling).

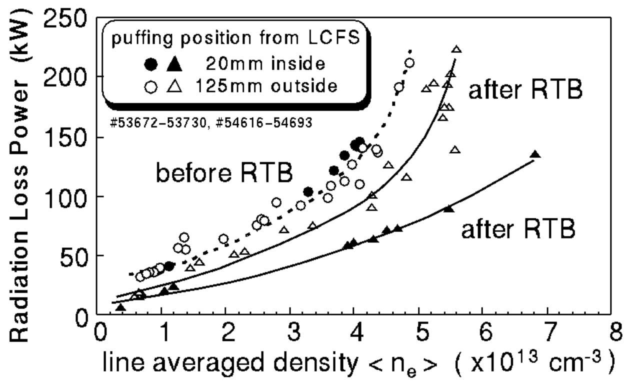

Fig. 3. Comparison of RTB effectiveness with injections from 20 mm inside and 125 mm outside the LCFS, where the total mass of decaborane used was 53 mg in both cases.

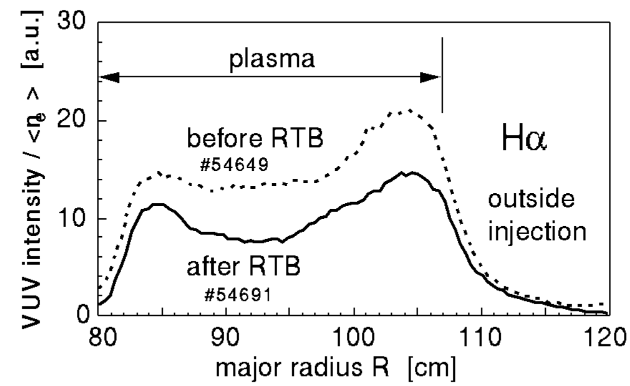

Fig. 4. The poloidal distributions of the H_α intensity before and after RTB with outside (125-mm) injection.

LHD scaling. Suppression of other impurities (C, Ti, and Fe) was also spectroscopically observed in visible or vacuum ultraviolet (VUV) light measured at a position toroidally opposite (180° away from) the RTB device.

Judging from the data points in Fig. 2, which were taken at the peak of line-averaged density in each shot by changing the H_2 gas puffing rate shot by shot, the lifetime of good conditions after RTB is longer than 10 shots — in other words, comparable with the total RTB time.

The most important result of this work is that the total puffing amount of decaborane was only 53 mg in 30 shots, which is equivalent to an average film thickness of 2 nm. This amount is almost 2 orders of magnitude smaller than that used in conventional methods such as glow discharge [3], reducing concern about flake dusts of coated films and hydrogen recycling. This result agrees with that obtained in TEXTOR using $B(CH_3)_3$ [4] and can be explained by assuming the coating of “preferential” or “wetted” small areas [5], such as the inner side wall of the chamber interacting with the LCFS as shown in Fig. 1, for instance.

Dependence on the puffing position

Figure 3 shows that the radiation loss power was more efficiently suppressed with inside rather than outside injection, while the total consumed amount of decaborane was 53 mg in both cases. It is conjectured that boride gases injected into the LCFS might be highly decomposed and ionized because of the high heating power and then distributed far into the torus along the magnetic field lines, thus avoiding local deposition near the puffing nozzle.

Effects on hydrogen recycling

As shown in Fig. 2, a low-density discharge sustained with wall fueling and without H_2 puffing was still reproducible even after RTB at room temperature, suggesting that the hydrogen recycling was comparable to that before RTB. Moreover, even with H_2 puffing at the same rate, the H_α intensity is smaller after RTB than before, indicating reduction of hydrogen recycling, as shown in Fig. 4.

To understand the present result, it is helpful to note the two advantages of using RTB. One is the extremely small amount of introduced decaborane. Also, the wall recycling is improved when increased power is used for the boronization. RTB needs no special power source because it uses the same power that is employed for plasma heating. According to simulation experiments [6], as the input power is increased when boronization is used, the hydrogen content in the boron films eventually decreases even at room temperature, probably because of enhanced dissociation of boride molecules, leading to efficient evacuation of H_2 gas products.

In conclusion, from the viewpoints of effectiveness, reproducibility, and controllability, the gas puffing method of RTB is quite promising as a wall-conditioning method to be used in LHD under the steady-state high magnetic fields produced by the superconducting magnet system.

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High-beta experiments at W7-AS

Upgrading of the neutral beam NB) heating power for Wendelstein VII-AS (W7-AS) in 1995 gave access to higher plasma energy contents and consequently higher β values. Therefore a major experimental aim was to test achievable β values and compare them with theoretically predicted β limits. The present contribution is based on Ref. [1] and concerns stationary or slowly varying plasma parameters.

Discharges in W7-AS with high NB heating power are characterized by a rapid density increase. Very high densities (up to 2.5×10^{14} cm^{-3}) can be reached; the maximum energy content and also the time behavior of the discharge are quite often determined by impurity radiation. Whenever the radiation exceeds about half of the heating power, the discharge is quenched at an energy confinement time scale. To some extent the radiation and also the density rise can be reduced by strong external gas puffing. A still slower density rise is observed after He glow discharges. This effect gradually vanishes. The highest β values are obtained at reduced magnetic field. Initial high-beta experiments in W7-AS were performed at B =1.25 T with electron cyclotron resonance heating (ECRH) plasma startup. Recent improvements of the 900-MHz nonresonant plasma start-up system made other magnetic field values available. Various initial ι values and stationary vertical fields were investigated. The plasma radius a_p at startup was determined by the inboard limiters via the vertical field and the ι value; at increased β the effective plasma radius is raised owing to the Shafranov shift.

Figure 1 shows as an example the development of a discharge at B = 1.25 T and ι = 0.43, yielding during a continuous density increase a nearly stationary plasma energy of ~16 kJ for more than 50 ms at full NB power. Figure 2 presents (a) radial electron density and (b) temperature profiles for a similar discharge, at a larger vertical field, a central density of about 2×10^{14} cm^{-3}, and a central electron temperature around 350 eV. Owing to the high density, the ion temperature should not be very different (no measurements are available at present). For this discharge, a central β of about 4% and an average β of between 1.7 and 1.8% are estimated in Fig. 2(c) by the free-boundary equilibrium code, NEMEC. The electron part of the pressure profile is taken from Thomson scattering data. The ion contribution is adjusted to match the measured diamagnetic signal. Internal toroidal net currents (bootstrap, ohmic, and Okhawa current) have been neglected because of the low electron temperatures. These β values approach the earlier predicted β limits of W7-AS of ~2%.

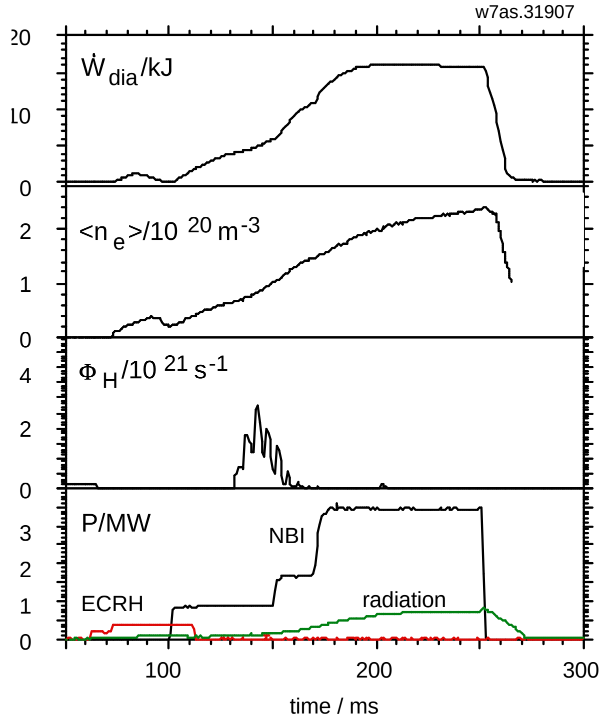

Fig. 1. Development of discharge 31907. From top to bottom: Diamagnetic energy, line-averaged density, external gas flux, and ECRH and NBI power input and bolometer radiation signals, respectively. (Courtesy R. Brakel)

The finite-β position of the magnetic axis and the plasma shape can be obtained experimentally from Thomson scattering (electron part of the pressure profile) and from surfaces of constant soft-X-ray intensity (evaluated on W7-AS by two soft-X-ray cameras via tomographic reconstruction), as shown in Fig. 3 for the "triangular" plasma cross section versus major radius coordinates, R, in comparison with calculated equilibrium flux surfaces by the NEMEC code. The code result is given in the top part of Fig. 4 for the two typical toroidal planes of W7-AS in comparison to the vacuum field and also yields the transformation between spatial and flux coordinates. The relatively large Shafranov shift of the magnetic axis at finite β leads to a considerable compression of the flux surfaces. This effect is especially pronounced at the outboard part of the "elliptical" plasma cross section. High local pressure gradients enhance plasma losses and may lead to instabilities. The bottom part of the figure shows the unstable region due to resistive interchange near the plasma edge as well as the radial profiles of β and ι. Stability studies at W7-AS are an ongoing field of investigation, especially in their correlation with MHD modes (see, e.g., Refs. [1–3]).

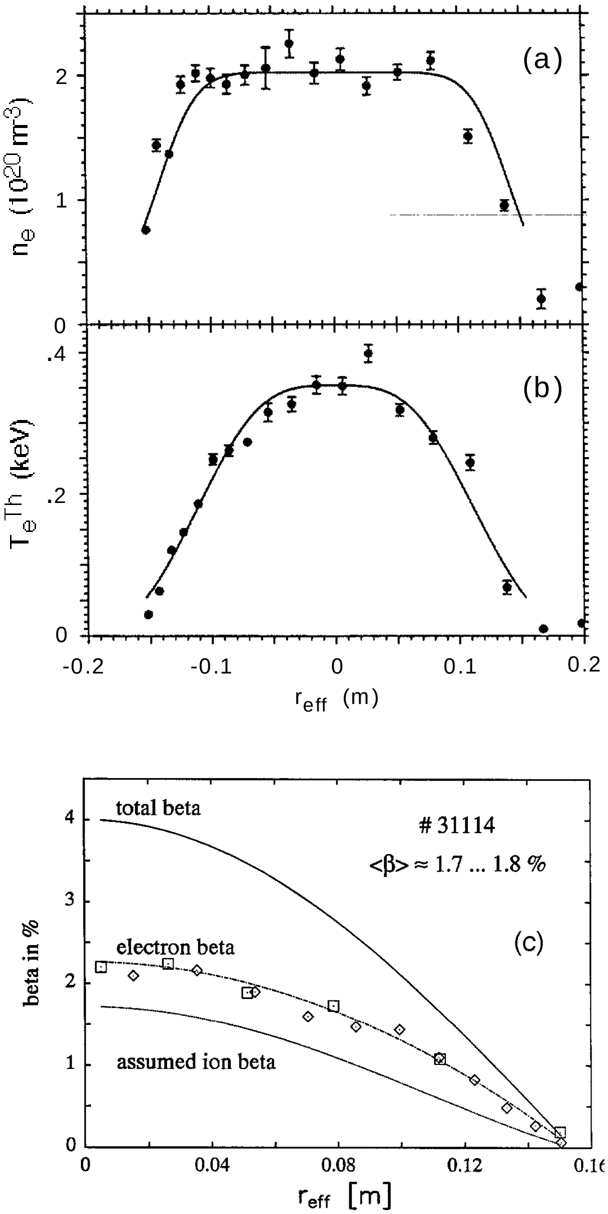

Fig. 2. Profiles of (a**)** Density, and (b) electron temperature, and (c) beta for a high-beta discharge (#31114) at B = 1.25 T and P_{NB} = 2.2 MW, yielding W_{dia} = 12 kJ, as a function of average minor radius r_{eff}. The density and electron temperature profiles were measured by Thomson scattering. For the beta-profile, the ion contribution was estimated. ⟨β⟩ is between 1.7 and 1.8%, depending on the exact contribution by the ions and the exact value of the plasma radius a_p. (Courtesy G. Kühner and J. Geiger)

The NB heating power and the rotational transform were varied in order to see whether a limitation in global confinement can be observed. In fact, the diamagnetic energy content does not linearly increase with heating power up to the highest available power. However, such a nonlinear dependence has to be discriminated from the usual power degradation of the confinement.

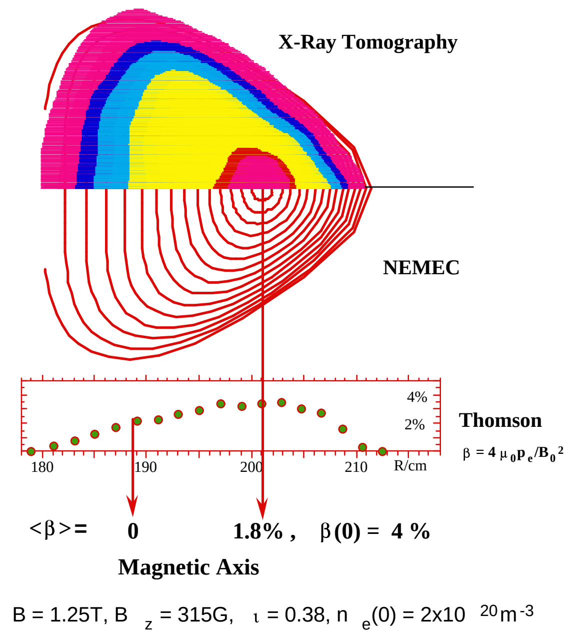

Fig. 3. Calculated finite-β flux surfaces of discharge #31114 in the 'triangular' plasma cross-section are compared with surfaces of constant soft X-ray intensity obtained by tomographic reconstruction of the signals of two cameras in the top part, and to the value of the electron-β measured along the major radius by Thomson scattering in the bottom part. (Courtesy J. Geiger, G. Kühner, and A. Weller)

Power scans at B = 1.25 T and at 2.5 T, where β differs by almost a factor of 2, show saturation effects which are too similar to allow a definite distinction in a preliminary analysis. In Fig. 5 the energy content of discharges with maximum NB heating power and constant line density is plotted as a function of the rotational transform. The Shafranov shift increases with decreasing ι in stellarators. Since at finite β the plasma is shifted away from the inboard limiters of W7-AS, the correlated growing plasma radius can roughly explain the increasing energy content according to $\tau_E \propto a^2$ if we assume otherwise constant transport properties. At the lowest ι values a slight saturation of the energy content can be seen. However, average β values (⟨β⟩ ≈1.6 – 1.7%) are rather constant during this ι scan and slightly lower than in the discharge presented in the previous figures. The measured MHD activity does not indicate any strongly confinement-degrading instabilities.

In summary: After the number of NB sources on W7-AS was doubled, the available higher heating power was used to investigate β limits. In these experiments a central beta of 4% and an averaged beta ⟨β⟩ of 1.7 to 1.8% were achieved. The stability code predicts resistive interchange modes to be unstable near the plasma edge. Nevertheless, fluctuation measurements as well as global plasma parameters do not clearly indicate a β limit so far. This

High-Beta Discharge 31114: Equilibrium and Stability

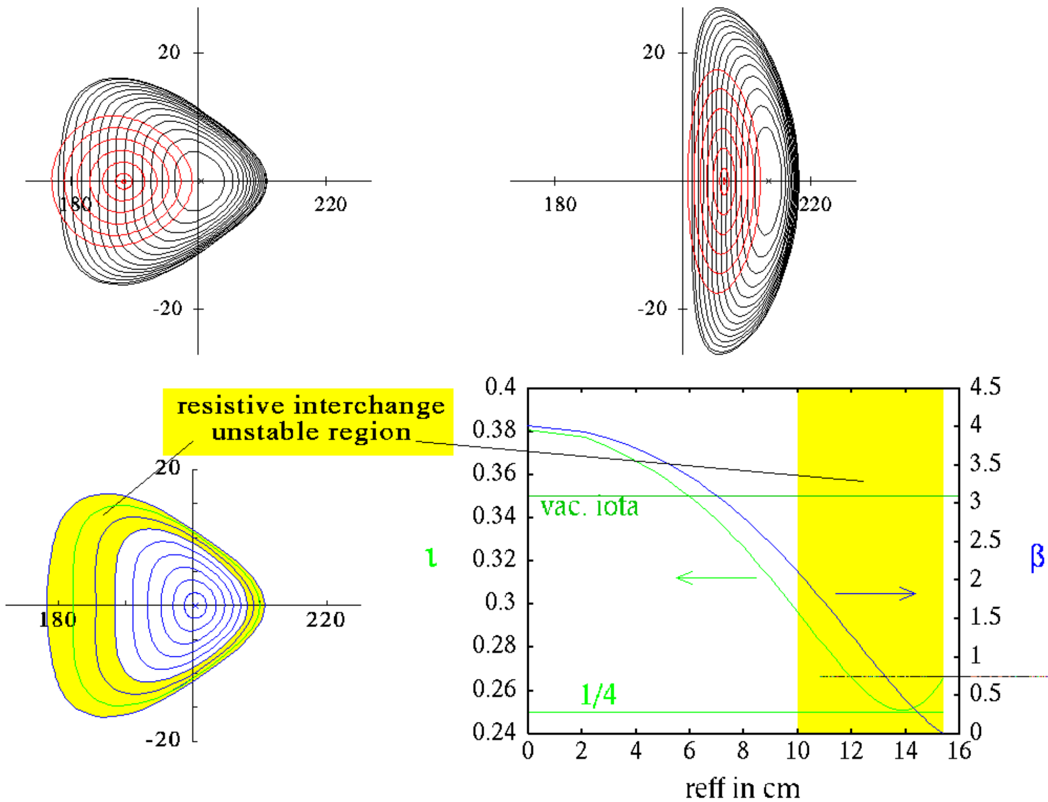

Fig. 4. Vacuum field and finite-β equilibrium flux surfaces calculated by the NEMEC code (top) and stability analysis with the JMC code (bottom) for the discharge conditions of Fig. 3. The plasma radius is set by the inboard limiters and expands at finite β as a consequence of the Shafranov shift. Distances are given in centimeters. The Shafranov shift in W7-AS is reduced compared to that in a "classical" stellarator, as expected from the reduction of the Pfirsch-Schlüter currents. (Courtesy J. Geiger)

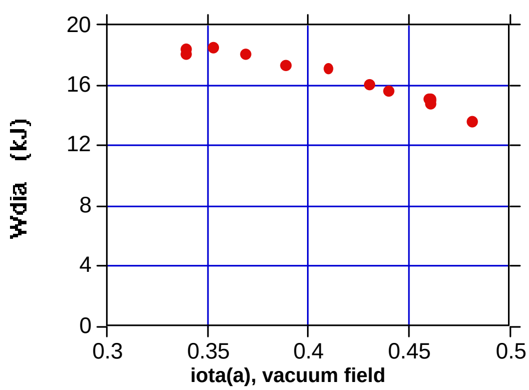

Fig. 5. The diamagnetic energy content of discharges 31900–31912 and 31914 for maximum NB heating power as a function of the vacuum field rotational transform ι at the plasma boundary. A maximum value of W_{dia} = 18.5 kJ is seen in discharge 31911 at ι = 0.353. The discharge conditions in this series are similar to those in Fig. 2 (ι = 0.38), except for the smaller vertical field of about 200 G. (R. Jaenicke)

implies that the experimentally achieved β values are essentially still power or radiation limited. The high central β values lead to a considerable Shafranov shift of the flux surfaces and consequently to high local pressure gradients. Possible problems with the equilibrium β limit are easy to avoid since W7-AS can be operated at higher rotational transform; on a fully optimized stellarator such as W7-X the shift of the flux surfaces as compared with the vacuum field remains small even at an average β of 5%.

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Method of magnetic analysis and plasma control for stellarators

Magnetic measurements to determine the shape and position of the plasma boundary together with the current density and plasma pressure profiles are widely used in modern tokamak experiments. However, in stellarators the situation is quite different. Usually, three integral quantities are measured in modern experiments: the total toroidal current I_p, the change in the averaged poloidal flux $\Delta\psi_p$, and the change in the toroidal magnetic flux $\Delta\phi_T$. These quantities are weak functions of the plasma profiles, and under conventional experimental conditions they provide no way to determine anything other than $\bar{\beta}$, where β is the ratio of the plasma pressure to the pressure of the magnetic field and the bar denotes the volume average.

Surprisingly, no attempts have been made to create a diagnostic system as effective as that for tokamaks. However, in analyzing the general properties of plasma-induced magnetic fields [1–3] we found no lack of components that are strongly dependent on the plasma profiles. Thus, we declared that the problem of magnetic diagnostics can, in principle, be solved [1, 2]. However, such declarations do not address the question of the feasibility of this method. Therefore, we decided to remedy this defect and following the fairly improvident "in for a penny, in for a pound" principle, performed an analysis of the problem. Naturally, it is impossible to give here the final solution. Recall that the theoretical basis of this problem in the case of more simple tokamaks has taken several years to develop and its different aspects were published in several tens of articles. Thus we shall restrict our consideration to what we consider to be the paramount problems. We have analyzed a way to distinguish two closely related peaked pressure profiles typical for modern experiments and to extract information on the distribution of the moderate current (for example, the bootstrap current or the current caused by high-frequency heating).

We demonstrate the general principles by considering a concrete example. We can claim *a priori* that a diagnostic system for stellarators is more complicated than that for tokamaks, and would therefore be justified only for a fairly large device. Among existing and forthcoming installations the choice is limited to two systems LHD (Japan) and W-7X (Germany). Taking into account the laboriousness of the problem and the capabilities of our computers we selected LHD.

LHD [4] is a torsatron/heliotron system with a large magnetic shear; the number of helical field periods $N = 10$ and the poloidal multipolarity $l_0 = 2$. We have considered here the case of ideal compensation; that is, we assumed that all the axisymmetric vacuum components produced by current-carrying three-dimensional (3-D) conductors are suppressed up to zero order. In this approximation the vacuum rotational transform ι_{vac} takes the value 0.4 at the magnetic axis and 1.2 at the separatrix, the aspect ratio $A = 7.3$. The most typical feature of LHD equilibria is the rather strong destruction of the plasma boundary at high β that was first investigated in Ref. [5]. We used the asymptotic iterative procedure developed in Ref. [6] to investigate this phenomenon.

In reality, the plasma-induced external magnetic field is a mixture of dissimilar harmonics (axisymmetrical and 3-D), and only a minor part of them is useful for our purposes. Genrally speaking, the extraction of information about the plasma profiles is connected with the resolution of plasma-induced harmonics higher than the dipole moment. Therefore, magnetic probes must be located fairly close to the plasma boundary. If a divertor is taken into account, it is impossible to use large averaging probes [2], and local probes must be used instead. Clearly, infinitely many probe systems satisfy these requirements. We are interested only in those with the minimal number of elements that can adequately measure the most important field components and specify the use of fast mathematical methods for the interpretation of magnetic data. We have proposed and analyzed five of ten different methods of magnetic analysis [7] and have chosen the most appropriate, which we call the *method of independent measurements*.

This approach uses two sets of magnetic probes that separately measure the poloidal and the longitudinal magnetic field components. Initially, the probes that measure the longitudinal component of the magnetic field are taken into account. All significant 3-D components are restored from these data using the set of different Legendre functions. For the sake of definiteness we have assumed a relative random error at the probes of the order of 5%, as is used typically in tokamak theory. For the analysis of the poloidal magnetic field, we used magnetic probes that independently measure two perpendicular poloidal magnetic field components (i.e., each magnetic probe measures the magnetic flux through the two perpendicular surface elements). The measured signal contains information about both two-dimensional (2-D) and 3-D field components. In the procedure of 2-D field restoration, we use 3-D field components restored previously.

We found there is a realistic possibility of determining the pressure profile in the framework of at least a one-parametric set of functions in the low-β case and in the framework of a two-parametric set of functions in the high-β ($\bar{\beta} > 1\%$) case. Also, it is quite simple to distinguish peaked and "bootstrap"-like current distributions. These satisfactory results can be achieved, for example, with the help of a magnetic probe system containing 20 poloidal probes in 2 cross sections and 24 toroidal probes in 3 cross sections. All the probes are located no closer than one-third of the average plasma radius to the outermost separatrix existing in the considered interval of $\bar{\beta}$ values.

Our calculations also showed that it is possible to optimize the accuracy of measurements by varying the number of probes and their distance from the plasma boundary. Technical recommendations for optimizing the accuracy of measurements if the total number of probes and the minimal distance from the plasma boundary for their location are specified are presented in Ref. [7]. Such measurements may form a basis for plasma control in stellarators. For example, the plasma boundary and the plasma pressure and current distributions can be determined in real time using previously prepared plasma equilibrium data. In particular, it is important for the considered device (LHD) whose characteristic feature is a rather strong destruction of the plasma boundary at high plasma pressures that require the use of external fields (e.g., vertical and quadrupole) for its restoration. However, these fields must be used with caution, since they may cause a degradation in plasma stability depending on the current and pressure distributions.

Most of the results concerning the principles of organizing probe systems are general in nature. However, it is pertinent to note that stellarators form a broad variety of confinement systems, and each of them may have distinctive equilibrium properties that must be taken into account. For example, as one might suspect by analogy with tokamaks with nearly rounded magnetic surfaces, the dipole component in LHD can be used only to determine the $\bar{\beta}$ value. However, our calculations demonstrated that the inward shift (due to, e.g., the external vertical field) not only causes a reduction of Pfirsch-Schlüter currents due to the magnetic hill increase, but also makes it possible to distinguish the pressure profiles considered in measuring the dipole component.

Another characteristic feature of LHD equilibria is a rather pronounced dependence of 3-D plasma-induced components on the pressure profile in the zero net current regime of operation, which provides a way to improve the accuracy of pressure profile determination in low-β cases. However, such behavior is not typical for the majority of confinement systems with larger aspect ratios. Our calculations showed that it is quite easy to distinguish peaked and “bootstrap”-like current profiles, but significant difficulties arise in distinguishing among current distributions that are similar at the edge but different in the center of the plasma column. At the same time, the latter problem is alleviated in systems with vertical elongation.

The number of probes required for such a technique is two- to three-fold larger than is needed for the equivalent tokamak. A simpler probe system containing about half as many probes as the general system can provide a sufficiently accurate determination of the plasma boundary and a rough estimate of the pressure profile.

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