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A first look at the confinement characteristics of NBI plasmas in LHD

The Large Helical Device (LHD) has significantly widened the operational regime available in previous helical devices. The understanding of currentless plasmas has been advanced in medium-sized stellarators such as CHS, Heliotron E, Wendelstein VII-AS, and ATF [1]. Confinement in these devices is comparable to that in L-mode plasmas in tokamaks with similar dimensions. Clear evidence of further confinement improvement has also been observed [2–4], most recently in W7-AS as seen in Fig. 1. However, the confinement enhancement factor has been smaller than in large tokamaks. This can be attributed to the influence of the boundary, where neutrals and atomic processes play an essential role, on the core where high-temperature plasmas are contained. LHD is large enough that it may exclude such complicated conditions. Neoclassical transport properties should become more evident in the radial transport in large-scale helical systems.

Four major inter-machine scalings of energy confinement time τ_E for helical devices have been proposed: the LHD scaling [5], the Lackner-Gottardi scaling (L-G) [6], the gyro-reduced Bohm scaling [7] and the 1995 International Stellarator Scaling (ISS95) [8]. All describe similar trends of τ_E deterioration with power and apparent positive density dependences. The ISS95 expresses

$$\tau_E = 0.079 a^{2.21} R^{0.65} P^{-0.59} \bar{n}_e^{0.51} B_t^{0.83} (\iota_{2/3})^{0.4},$$

where τ_E is in seconds, a and R in meters, P in megawatts, $\bar{n}_e$ in $10^{19} \mathrm{m}^{-3}$, and B_t in tesla. $\iota_{2/3}$ is the rotational transform at the two-thirds radius. While this scaling also describes L-mode plasmas in large tokamaks, the density dependence in helical devices is pronounced, in contrast to the tokamak L-mode scaling.

Experience in tokamaks shows that, larger devices, which have lower heating power density, experience a saturation in τ_E from linear ohmic confinement in a fairly low density regime; this is connected to the appearance of an L-mode with auxiliary heating. This issue is crucial for the prospects of helical devices because the present reactor designs rely upon the favorable density dependence of confinement. Because the confinement of ELMy H-modes suggests a similar density dependence, a better understanding will be gained by comparing LHD with large tokamaks.

Figure 1 shows experimental data on τ_E and the predictions of the ISS95 expression. NBI-heated plasmas in the second LHD experimental campaign have achieved the target area and are comparable to large tokamaks in terms of energy confinement time. The density and power absorption ranges cover $(1.0–5.0) \times 10^{19}$ m^{-3} and (0.5–2.3) MW. The

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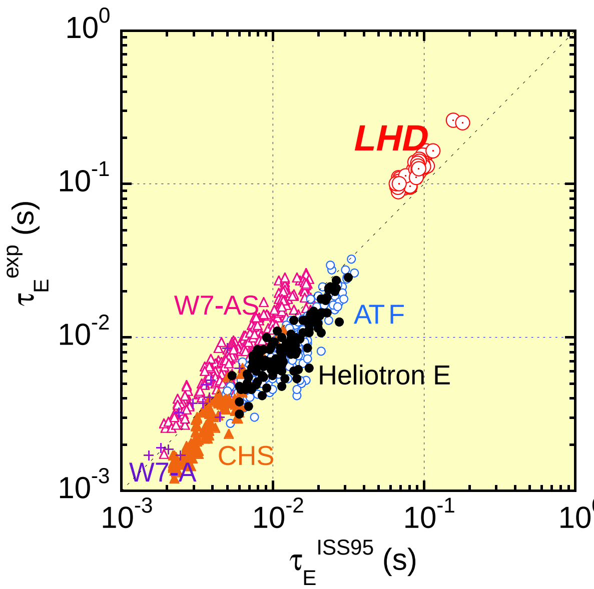

Fig. 1. Comparison of energy confinement times obtained in experiments with the ISS95 scaling.

magnetic field is 1.5 to 2.75 T. Data from hydrogen discharges are presented here, since at this point, the estimated power deposition in helium discharges cannot be discussed with the same confidence as that in hydrogen. Data from NBI-heated plasmas are systematically shifted upward from the prediction line, which indicates enhancement of confinement by about 40–50% compared with ISS95. This deviation from the scaling indicates an enhancement of a factor of 2 from the LHD scaling, comparable to the L-G scaling.

A regression analysis of the LHD data gives the best-fit expression $\bar{n}_e^{\,0.4}\, P^{-0.6}$, which is similar to prior scaling laws, indicating a favorable density dependence and power degradation. Under the present experimental conditions, however, the absorbed NBI power depends on the operational density. Therefore careful attention and further systematic scans are needed to quantitatively evaluate the power degradation from a favorable density dependence with sufficient accuracy. High-density plasmas with pellet injection and one neutral beam injector have almost the same absorption power as low-density plasmas with two injectors. Comparison of these sequences indicates that energy confinement improves with the square root of the density in the accessible density range (3×10^{19} m^{-3}) with a power density of 40 kW/m^3. This is despite the prediction of confinement saturation at 1.7×10^{19} m^{-3} according to the tokamak experience of transition from linear ohmic confinement to a saturated ohmic confinement regime. A simple statistical analysis including data from LHD has not provided an expression consistent with theoretical dimensional analysis constraints. Revision of the scaling law would require introduction of some non-dimensional parameters.

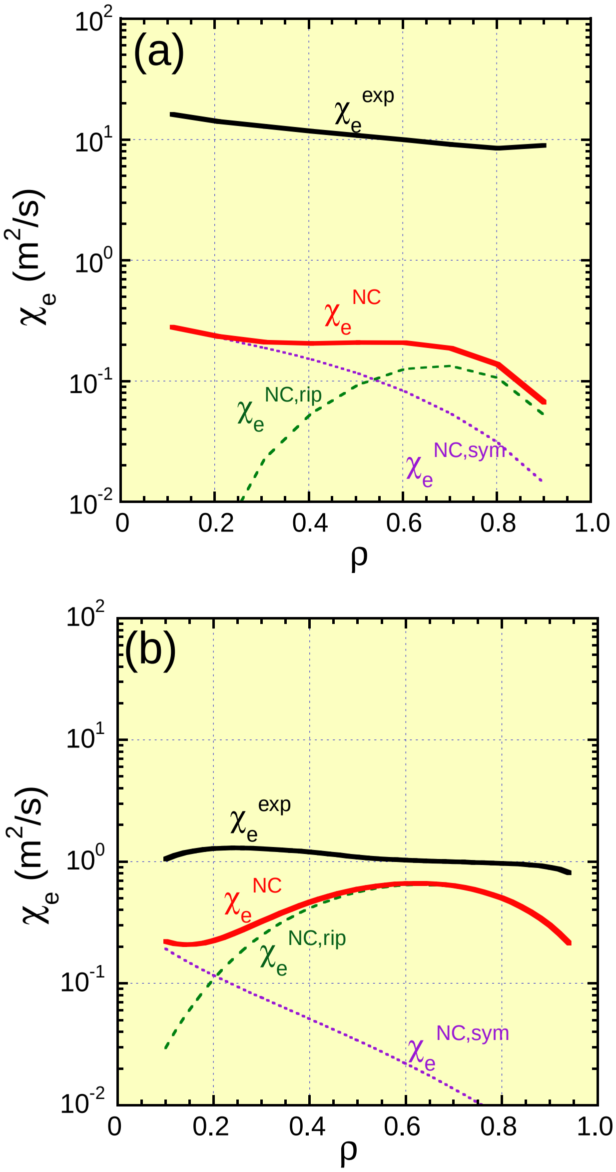

Fig. 2. Radial profile of electron heat conduction coefficients χ_e in (a) CHS and (b) LHD. Superscripts denote experimental values (exp), neoclassical values (NC), and contributions from toroidal ripple (sym) and helical ripple (rip). The collisionality is almost the same for both cases. The normalized gyroradii in the case of LHD are smaller than in the case of CHS by a factor of 2.5.

One-dimensional heat transport analysis is proceeding on the basis of a variety of profile measurements. Figure 2 shows an example of an electron heat conduction coefficient χ_e derived from experiments in CHS and LHD. Since collisionality (the ratio of collision frequency to the bounce frequency of helically trapped particles) is around 0.3–0.7, collisionless helical plasma characteristics are pronounced. The heat conduction of electrons in LHD is around 3 as large as that predicted from the neoclassical theory. However, this anomaly is much smaller than that observed in medium-sized helical plasmas (CHS, *R/a* = 1 m/0.2 m) with the same collisionality. This reduction in anomalous transport may not be explained by the reduction of the ratio of poloidal gyroradii to the plasma minor radius. This improvement in electron heat transport may be one of the keys to the enhancement of LHD confinement over the prior scalings.

Regarding ion heat transport, experimental data appears to be consistent with the neoclassical model employing an electric potential with the same value as the temperature. This supports a transition from the 1/ν regime to ν regime and strongly suggests that helical ion ripple transport can be mitigated by the radial electric field.

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Temperature pedestal discharges in LHD

Enhancement of τ_E by edge control is one of the main physics goals in the LHD program. During the design phase, we proposed high-temperature divertor operation, in which the edge temperature is raised by efficient pumping, leading to enhancement of τ_E [1]. We are now developing a pumping system to achieve this.

In the first and second campaigns, we cleaned the vacuum vessel wall to such an extent that fairly strong wall pumping appeared during the discharge. Under these wall conditions we observed a clear pedestal at the edge of the T_e profile. T_e profiles were measured by Thomson scattering along the major radius (R) at the poloidal plane where the plasma is elongated horizontally (Fig. 1c). In Fig. 1a, the T_e profile during the flat top of a typical LHD discharge is plotted as a function of R. A mild pedestal with a shoulder temperature (temperature at $\rho = 0.9$, T_e^{ped}) of ~500 eV can

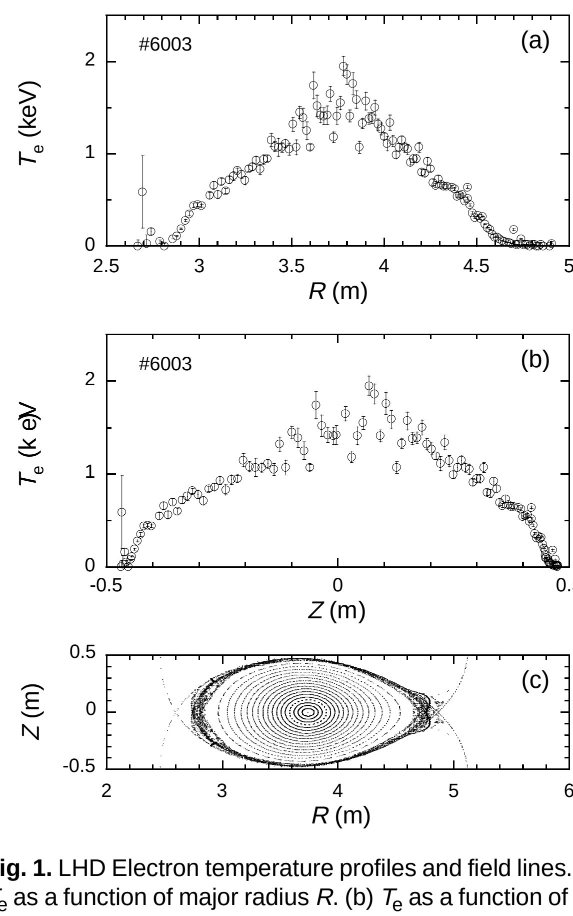

Fig. 1. LHD Electron temperature profiles and field lines. (a) T_e as a function of major radius R. (b) T_e as a function of the vertical axis Z; data points for positive Z are from the outer region (R > 3.75 m), while those for negative Z are from the inner region (R < 3.75m), because of the rotation of the field lines. (c) Puncture plot of the field line for the LHD configuration.

be seen. In the edge region where the pedestal exists, the flux surface is significantly expanded toward the X-points (Fig. 1c). Thus it is more appropriate to plot it as a function of Z (the vertical axis) with an assumption that T_e is constant within the flux surface (Fig. 1b). In this plot, a much sharper T_e gradient appears at the edge. The width of the pedestal is ~4 cm in the Z direction, which appears to be larger than the poloidal ion gyroradius (1.5 cm). The maximum shoulder temperature obtained so far is 650 eV. This is very high, considering that it is typically 70–80% of the average temperature. Thus T_e^{ped} determines W_p and hence τ_E.

The LHD edge magnetic structure is complicated. The outward heat and particle fluxes that cross the last closed flux surface (LCFS) pass through an ergodic region and then a more open "edge surface layer" before reaching the X-point and the divertor plates. We have found that ∇T_e is negligibly small in the clearly open ergodic region and the temperature is below 50 eV. On the other hand, a significant density gradient exists in the open region and the density at R = 4.65 m, where T_e starts to rise, is about 50% of the shoulder density, which is close to the average density. Particles appear to be confined in the open ergodic region. This is not surprising since cold ions are well confined in the edge region where the connection length is longer than ~300 m.

The pedestal in the LHD discharge forms during the rising phase, not through a rapid transition. Figure 2 shows the temporal evolution of an LHD discharge. Starting at 400 ms, a neutral beam (3.1 MW) is injected into a small ECRH-generated target plasma discharge ($n_e \sim 1 \times 10^{19}$ m^{-3}, central temperature $T_e^0 \sim$ 300 eV) which has cold plasma (below 50 eV) in the outer region ($\rho > 0.6$). With beam heating on, the hot plasma region expands radially and eventually reaches the LCFS, as evidenced by an increase in T_e^{ped}. After 10–20 ms, T_e^{ped} becomes 200 eV. Simultaneously, hot plasma also reaches the divertor plates, as shown by increase in the ion saturation current (Γ_{div}) measured by the probes at the inner divertor plate. At t = 600 ms, the gas puff is turned off, and 100 ms later, the density starts to decrease. But W_p increases by ~30%, accompanied by a substantial increase in T_e^{ped}. During this evolution, ($T_e^0 - T_e^{ped}$) remains fairly constant, and thus increasing T_e^{ped} appears to lead to higher W_p and longer τ_E. Normally a decrease in the density accompanies that in W_p in this density range. This result demonstrates that even with the same external global parameters (n_e, B, input power), the energy confinement differs depending upon the value of T_e^{ped}.

In the tokamak H-mode, edge-localized modes (ELMs) appear repetitively, expelling particles and energy to the divertor plates. It has been argued that ELMs are MHD phenomena caused by ballooning modes, which become

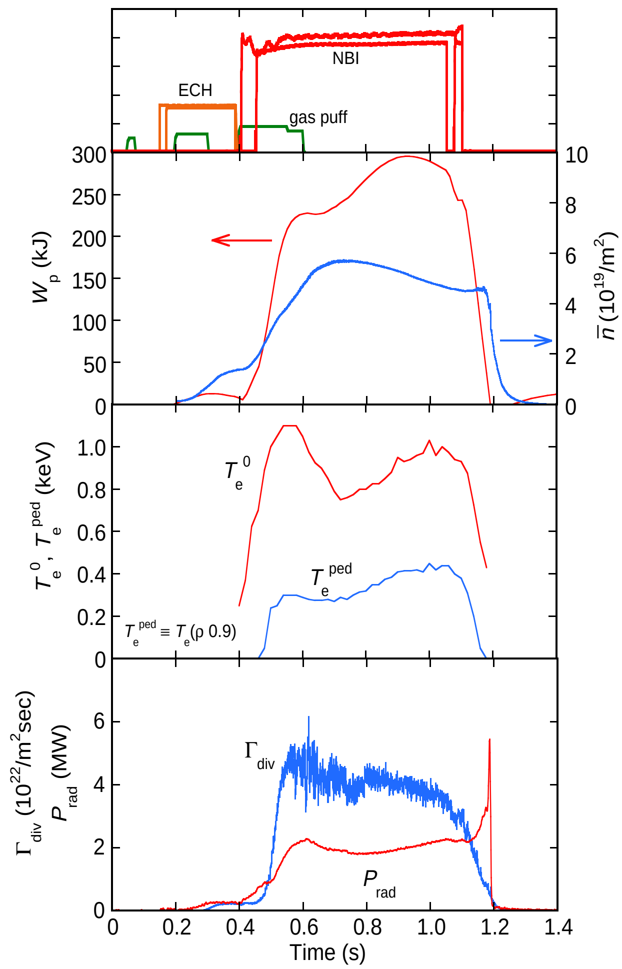

Fig. 2. Temporal evolution of an LHD discharge that exhibits a temperature pedestal.

unstable when pressure gradient exceeds a critical value. In the LHD discharges, we have not seen such relaxation phenomena. But the edge temperature and density at the pedestal shoulder are comparable to those of medium sized H-mode tokamaks, such as ASDEX-Upgrade, and the normalized pressure gradient, defined as $\nabla p_N = a\nabla P/(B^2/2\mu)$, is as high as 0.07 for the discharge shown in Fig. 2.

The sharp temperature gradient normally appears at the edge. For the LHD configurations used in the experiment, the ɫ =1 surface and a small m/n = 1/1 island generated by an error field are located in the edge ($0.8 < \rho < 1.05$), and thus the location of the high ∇T_e region can be interpreted to be near the ɫ =1 surface. In LHD low-density operation (B = 2.5 T), a pedestal with ∇T_e as high as ~100 eV/cm also appears in the core ($\rho < 0.8$) only during the plasma expanding phase of the discharge. But we also note that the gradient region is close to the ɫ = 1 surface (ρ = 0.9). We cannot eliminate the possibility that the m/n = 1/1 island or the ɫ =1 surface plays some role in the formation of the pedestal. It is interesting to note that in smaller helical devices, such as CHS and W7-AS, the H-mode has been observed only when the LCFS is a major rational surface.

We imposed an $m/n = 1/1$ island on the standard configuration and studied its effect on the plasma. In the temperature profile of Fig. 3, flattening of the T_e profile around the $\iota\!\!\!-$ = 1 surface is clearly seen. When the island size is increased, the pedestal is gradually destroyed and hence τ_E decreases accordingly. We also observed flattening of the density profile due to the island. One of the ideas behind using the island geometry is that suppression of the density at the LCFS for values well below 1×10^{19} m^{-3}, leads to the generation of a sharp density gradient just inside the LCFS and τ_E enhancement, but we failed to suppress the density. Further attempts are planned in a future experiment.

One may naturally ask whether the LHD discharges discussed here are a version of the H-mode observed in tokamaks and small helical devices. Similarity exist in the formation of the temperature pedestal, which leads to enhancement of the energy confinement. The LHD discharges do not exhibit many of the key features that characterize the H-mode, such as (1) L-H transition, (2) n_e pedestal and drop in recycling, and (3) edge relaxation phenomenon, ELM. More detailed measurements, including measurements of the potential (and its fluctuations) with a heavy ion probe, are planned to clarify the physics mechanism of pedestal formation. By substantially increasing the NBI heating power, we will also attempt to raise the pedestal temperature to a few kiloelectron volts to demonstrate high performance of the helical device.

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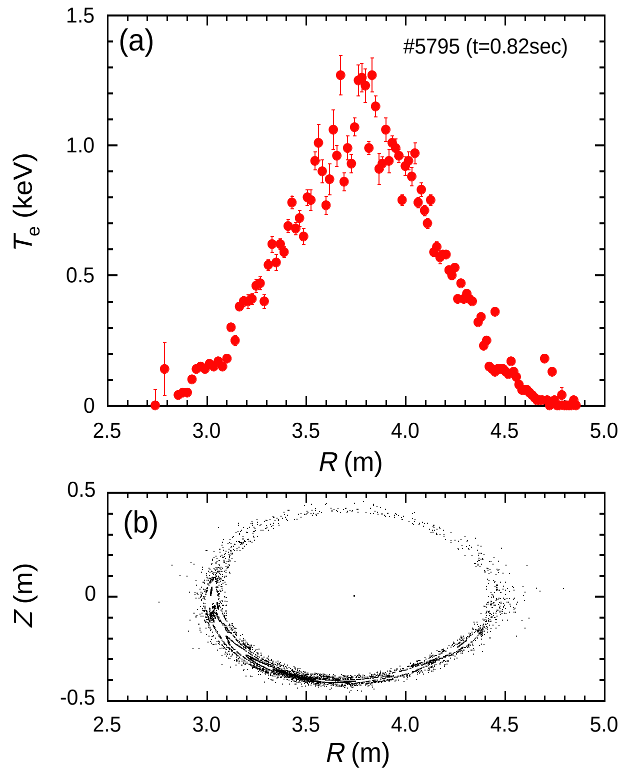

Fig. 3. (a) Flattening of T_e profile around the $\iota\!\!\!-$ = 1 surface by an Island (m/n = 1/1). (b) Puncture plots, showing existence of the island.

People

Upon the retirement of Professor Grieger

Professor Günter Grieger retired from the Max Planck Institut für Plasmaphysik on February 28, 1999. Prof. Grieger has been a strong leader of stellarator research at the Institute. He has also been a principal promoter of and a major contributor to world stellarator research. In the Wendelstein program, he initiated the Wendelstein-A and Wendelstein-B programs and led the experimental studies on these devices.

After many attempts and struggles to improve stellarator plasmas, he achieved remarkable results which eventually led to the construction of larger devices, W7-A and W7-AS. He is a true leader who continued to proclaim the potential advantages of the stellarator even in the days when many tokamaks were being constructed, one after another, following the disappointing pump-out phenomenon seen in the C-stellarator at Princeton in the 1960s. He continued stellarator research vigorously and overcame this obstacle in stellarator research.

Recently he has focused his efforts on the realization of the W7-X device, which is under construction in Greifswald. Prof. Grieger has been the chairman of the Stellarator Executive Committee and a leader in stellarator research for many years, giving constructive support and advice to other countries' programs. He also took on the responsibilities of Vice-Chairman and later Chairman of the IEA Fusion Power Coordinating Committee and thus has been a world fusion research leader, playing a major role in world fusion research activities. He has been instrumental in reaching many important agreements in various areas of fusion, leading to advances in tokamaks, stellarators, reversed-field pinches, reactor materials, and reactor engineering. It is with regret and a profound sense of loss that we acknowledge the retirement of Professor Grieger, a man with a great mind and a deep understanding of fusion plasma physics. We hope that he will continue to provide advice and guidance to us as he has in the past.

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Meetings

International Stellarator Workshop

The biennial International Stellarator Workshop will be held September 27 through October 1 at the Monona Terrace conference facility in Madison, Wisconsin. The previous workshop was held in Toki, Japan in September, 1997. Approximately 100–150 participants are expected because of the interest in stellarator research.

The conference will cover all aspects of helical systems research: overviews of major experiments; transport; MHD equilibrium and stability; plasma edge and turbulence; particle and power handling, divertors; plasma eating; diagnostics; configuration optimization; new devices; and reactor studies.

The conference will start Monday morning and end at noon on Friday, and will consist of both oral and poster presentations. Details on registration, housing reservations, travel, conference banquet, the format for the 4-page contributed papers, etc. will be available by the next issue of *Stellarator News*.

Contact Jim Lyon (lyonjf@fed.ornl.gov) or David Anderson (dtanders@facstaff.wisc.edu) for more information.

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