DPPL-CFP--3262
CONF-9409369--1

Magnetic Confinement Experiment - I: Tokamaks

Robert J. Goldston, Princeton Plasma Physics Laboratory

Introduction

Reports were presented at this conference of important advances in all the key areas of experimental tokamak physics: Core Plasma Physics, Divertor and Edge Physics, Heating and Current Drive, and Tokamak Concept Optimization. In the area of Core Plasma Physics, the biggest news was certainly the production of 9.2 MW of fusion power in the Tokamak Fusion Test Reactor, and the observation of unexpectedly favorable performance in DT plasmas. There were also very important advances in the performance of ELM-free H- (and VH-) mode plasmas and in quasi-steady-state ELM'y operation in JT-60U, JET, and DIII-D. In all three devices ELM-free H-modes achieved $nT\tau$'s ~2.5x greater than ELM'ing H-modes, but had not been sustained in quasi-steady-state. Important progress has been made on the understanding of the physical mechanism of the H-mode in DIII-D, and on the operating range in density for the H-mode in Compass and other devices.

In the area of Divertor and Edge Physics the major new advance is that pumped divertors are now nearly routine "tools of the trade," used for helium pumping, density control, and impurity control. Experiments on DIII-D demonstrated clear control of density, as well as helium pumping quite adequate for a reactor – in ELM'ing H-mode plasmas. The ASDEX-Upgrade team reported results in which feedback control of both deuterium and neon puffing allowed operation in a so-called "Completely Detached H-mode (CDH)." In this mode even energy dumps from ELMs did not burn through to the divertor plate. The first results from vertical-plate divertors in C-Mod and JET are very promising. The C-Mod team has also shown successful ICRH heating in an all-molybdenum machine.

In the area of Heating and Current Drive, there were many advances, but perhaps the clearest theme was the multiplicity of applications that have been found for ion-cyclotron-range-of-frequency RF power. Alfvén wave current drive has been demonstrated on Phaedrus-T; second harmonic tritium heating has been explored on TFTR, as has mode-conversion current drive. Ion Bernstein waves have been used to create a core transport barrier in PBX-M, and progress has been made on fast wave current drive on both Tore Supra and DIII-D. At higher frequencies, lower-hybrid current drive is now becoming a reliable and well-understood tool for current profile control in JT-60U and JET.

Tokamak Concept Optimization was an important theme of each of the other three key experimental areas, as well as a topic of discussion in its own right. For example, in the Core Plasma Physics sessions, high triangularity, low recycling plasmas in JET were shown to have the longest ELM-free H-modes and the highest performance. In the Divertor and Edge Physics sessions DIII-D results were presented indicating that a pumped divertor can be used to unload particles from the wall, presenting the possibility of very-low recycling, higher performance plasmas in steady-state tokamaks. Presentations were also made on how the current-profile-control tools that have been developed in previous years are now being applied to develop new "advanced-tokamak" regimes on JT-60U and JET. Specifically in the "Concept Optimization" sessions, new progress was reported on DIII-D in understanding the core physics of VH-modes, which may be tied to Mercier stability, and the role of plasma rotation in stabilizing wall-modes. Important results were reported on reversed-shear and high q(0) modes in TFTR and DIII-D, and new ground was broken by the START team in the development of low-aspect-ratio tokamaks. Important advances in disruption amelioration were reported from JT-60U, and results from ASDEX-Upgrade and Alcator C-Mod deepened our understanding of Vertical Displacement Events. Perhaps the crowning achievement in this arena is the attainment of high bootstrap fraction and full current drive in JT-60U, in a mode which could be prototypical of future steady-state reactors.

What can be learned from all these results? The "big picture" is that the tokamak is no longer the rigid, self-determining system of the early 1980's, with sawteeth in the core, high neutral recycling at the edge, L-mode confinement globally, and Troyon β limits. We can control the current profile, we can control the edge conditions, and with skill (and a little luck) we can coax the plasma to very high confinement and pressure. A question that remains is how much of this high performance we can harness to improve the attractiveness of the tokamak as a fusion power source.

Core Plasma Physics

A major highlight of this meeting was the report that 9.2 MW of fusion power had been produced in supershots in TFTR. This constitutes more than a factor of five improvement over the results reported from JET at the last conference in 1992. Wall conditioning with lithium pellets has dramatically reduced particle recycling and carbon influx, and has facilitated the extension of the enhanced-confinement "supershot" regime to the full range of currents accessible in TFTR. Now fusion power production in TFTR is more limited by gross MHD stability than by confinement. The very peaked pressure profiles in TFTR supershots lead to extremely high central fusion power densities (exceeding that expected in ITER), but also result in a lower-than-usual volume-averaged pressure limit, $\beta_N \leq 2$. A remarkable result in these high performance discharges is that the scaling of energy confinement with ion mass appears to be very strong, resulting in global stored energy ~20% higher in DT plasmas than in equivalent DD discharges. This is attributed largely to the ion channel, where a reduction in $\chi_i \propto \langle A_i \rangle^{-1.4}$ is observed.

In some of the highest power DT shots, enhancements have been observed in the Mirnov signals in the range of frequencies that could correspond to α-driven Toroidal Alfvén Eigenmodes (TAE's). The signal levels, however, were still lower than those observed to cause loss of beam ions during beam-driven TAE's, and indeed no enhanced α loss was observed. Absolutely calibrated charge-exchange recombination spectroscopy has been used to observe the slowing-down α's in the plasma in the energy range 80 – 800 keV. The measurements are in remarkably good agreement with Monte-Carlo-code cal-

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culations based on classical collisional thermalization. In cases with sawteeth, the central α density is somewhat depleted, as expected. After the α's slow down they are observed to transport across the plasma with $D \sim \chi_e$, very good news to dispel concerns about helium ash accumulation. All in all, these TFTR results have been uniformly positive. We should wish the TFTR team success in their plan to extend these high-fusion-power plasmas to pulse lengths in excess of 1 sec, in order to enhance the α parameters for further studies.

Another very important result reported at this conference was the commissioning of the "new JET," with internal divertor coils and a divertor cryopump. The first results on power-handling in the new configuration are extremely encouraging. With the use of divertor sweeping, up to 140 MJ have been passed through the plasma without exciting carbon "blooms." Quasi-steady-state ELM'ing H-modes have been established for 20 seconds, with good H-mode confinement. Unfortunately, the highest performance ELM-free hot-ion H-modes reported at the last conference have not yet been accessed in the new JET. Experimental results suggest that both increased recycling compared to the previous campaign, and the lower triangularity of the new configuration are contributing to this problem. Interestingly, however, in both the old JET, and in the new one, the $n_{do}T_{io}\tau_E$ attained in these ELM-free hot-ion H-modes (or VH-modes) is about 2.5x greater than can be achieved in ELM'ing H-modes. A very interesting new point in the JET data set is a 4 MA ELM-free H-mode with a density of nearly $10^{20}/m^3$, a high $n_{do}\tau_E \sim 4\ 10^{19}$, $T_{io} \sim 11$ keV, and $T_{eo} \sim 11$ keV.

A new world-record $n_{do}T_{io}\tau_E$ was reported in JT-60U in short ELM-free phases of "high-β_p H-modes." Very similarly to the JET results (old and new), the fusion triple product drops by a factor of 2.5 from transient ELM-free to quasi-steady ELM'y H-modes. The JT-60U team has been able to sustain the ELM'y H-mode in quasi-steady-state for 1.5 sec with up to 30 MW of heating power, before problems were observed with enhanced carbon influx. These results put JT-60U in the same range of $Pt^{1/2}$ (a figure of merit for surface heating) as JET.

The DIII-D team has done detailed shape scans, investigating the effect of elongation and triangularity on ELM-free H-modes. They find that high elongation, and especially high triangularity, are required for the highest performance. They further find that the best ELM-free VH-modes, though transient, attain about a factor of 2.5x higher $\langle nT\rangle\tau_E$ than ELM'y H-modes. (Note that this measure *does not* give credit for the high T_{io}/T_{eo} attained in hot-ion VH-modes, as does $n_{do}T_{io}\tau_E$.) Figure 1 shows the highest performance results from JET and JT-60U ELM-free H-modes, plotted against data from the DIII-D shape scan. The abscissa represents potential fusion power density, assuming that the attainable toroidal magnetic field at the plasma scales as $\varepsilon^{-1/2}$. β/ε also equals $\beta_N f$, where $f \equiv RI_p/(a^2B) \propto 1/q_{cir}$. The ordinate represents $\langle nT\rangle\tau_E$ (where τ_E is defined conservatively as W_{tot}/P_{tot}) normalized to a "generic" L/H-mode scaling: $\langle nT\rangle\tau \propto (I_p/\varepsilon)^2 \propto (aB)^2f^2$. Along both axes, then, success in maximizing f at fixed a, R, and B, by strong shaping or low q operation, shows up as improved performance, even at fixed H or β_N. Improved performance corresponds to higher fusion power density (along the abscissa) or a lower requirement for a^2B^2 to attain a given fusion gain, along the ordinate. An outline of the data contained in the ITER H-mode database, constrained to modest shaping, is shown in the lower left-hand corner. (Strongly shaped data from PBX-M show very high β/ε and $\beta\tau_E/a^2$, but have been constrained out of the H-mode dataset here.) The overall plot indicates that if steady-state, high performance ELM-free operation can be attained in a strongly shaped reactor plasma, it will be possible to reduce the size of a fusion reactor and increase the fusion power density considerably. Also shown on this plot are the minimum target operating points for ITER and TPX. Given their shapes, both planned devices reach somewhat beyond the established database. ITER adds the challenge of ignition and long-pulse burn, while TPX adds the challenge of steady-state operation.

A number of smaller devices contributed further insights on confinement scaling at this meeting. Both the TUMAN-3 and T-11 teams reported confinement in ohmically heated plasmas with boronized walls far above neo-Alcator scaling. At extreme low aspect ratio, START results also exceeded neo-Alcator scaling by a large margin. Perhaps the clearest indication of the demise of neo-Alcator scaling is the report from Alcator C-Mod that their ohmically and RF heated plasmas both fit well to L-mode scaling!

The DIII-D team has been pursuing more fundamental studies of confinement scaling by attempting to make "dimensionlessly similar" discharges, in which only the ratio of the gyroradius to the system size varies. In principle such discharges can "point" to ITER along a line of constant β and ν^*, but varying ρ/R. In detailed L-mode scans they found that the electrons behaved favorably, like gyro-Bohm scaling [$\chi \propto \chi_B(\rho/R)$], while the ions behaved less favorably than Bohm, approximately as $\chi_B(\rho/R)^{-1/2}$ (They termed this particularly unattractive behavior "Goldston" scaling.) Together these results were hypothesized to give rise to the overall Bohm-like L-mode scaling normally observed. Very curiously, in H-mode *both ions and electrons behaved like gyro-Bohm*. This is very difficult to understand, since the H-mode confinement enhancement factor is similar in JET and in JFT-2M, devices of radically different aB. On the other hand, there *are* indications of gyro-Bohm scaling in the H-mode database. Perhaps connected to this is the observation on JET that the effect of an edge H-mode transition propagates inward at a speed of 150m/sec. Geoff Cordey suggests that this may be the result of some large-scale (*i.e.*, Bohm-like) influence of the edge turbulence suddenly being turned off at the H-mode transition.

In the area of H-mode physics the DIII-D team presented data with very high time and space resolution near the plasma edge, at an H-mode transition, showing clearly that the radial electric field due to poloidal rotation as well as the turbulence suppression and D_α light drop, develop well before the electric fields due to toroidal rotation and ion pressure gradient build up. The timing of the poloidal rotation and turbulence suppression are such that it appears the rotation precedes the turbulence suppression, but this is more open to debate. The JT-60U group presented detailed data demonstrating the internal transport barrier in the ion channel that is observed in the "high β_p" mode. Both rotation shear and ion temperature gradients build up first in a region inside of q = 3, and then the steep gradient moves outwards to the vicinity of $q = 3$, and stops there.

A number of groups showed results on H-mode threshold power. The so-called ASDEX-Upgrade scaling of $P_{tot}/S = 0.044\ n_{e20}B_T$ holds fairly well over the full range of device sizes and fields from Compass to JET and JT-60U. A potentially troublesome result for ITER is that there also appears to be a *lower* density limit for H-mode access, as reported in the first experiments on the old ASDEX, and explored in further detail on Compass. The scaling of this lower limit is uncertain. Fritz Wagner suggested that if the H-mode is primarily associated with edge *ion* parameters, perhaps the weak electron-ion coupling at low density results in difficulty reaching the H-mode. Ion-electron coupling should be rather strong, even at low density, in ITER, due to the high value of τ_E.

Divertor and Edge Physics

This was the first year that extensive results from tokamaks with pumped divertors have been reported, and the results are very encouraging. The most exciting recent news is the achievement of "Completely Detached H-modes (CDH)" on ASDEX-Upgrade. The ASDEX-Upgrade team has found that it is possible, through a combination of deuterium and neon puffing, to sustain an H-mode plasma with as much as 90% radiated power (increased from 55% without neon puffing). This was achieved through feedback control of the D_2 puff to maintain a constant neutral pressure in the divertor, and feedback control of the neon puff to maintain a specified radiated power. This kind of feedback control is made possible by an effective pumping system, as was demonstrated earlier on TEXTOR using a pumped limiter. By radiating a large fraction of the injected power from just inside the separatrix, the ASDEX-Upgrade team was able to lower the power crossing the separatrix apparently to just above that needed to sustain the H-mode. Under these circumstances rapid type III ELMs are observed, which do not dump enough energy to burn through the radiating plasma to the divertor target plates.

This CDH mode may be attractive for use on ITER, but a number of problems remain to be solved. First, the use of neon as a radiating impurity may be prohibited by the associated fuel depletion. In ASDEX-Upgrade $\Delta Z_{eff} = 0.6$ for $\Delta P_{rad}/P_{aux} = 35\%$ @ $P_{aux}/S = 0.17$ MW/m^2, to be compared with ITER's $P_\alpha/S \geq 0.25$ MW/m^2. Fuel depletion increases both the $nT\tau$ required for high fusion gain, and also the β required to produce a given fusion output power. Perhaps a higher Z impurity would be more appropriate for ITER application. Second, it may be problematic that the anticipated H-mode threshold power for ITER, at its full operating density, is quite high. It is possible that a large fraction of ITER's α power will be required to sustain the H-mode, and therefore must be allowed to cross the separatrix into the SOL. Finally, although the H-mode confinement was not significantly degraded by the strong edge and SOL radiation, the overall H-factor in this regime was only in the range of ~1.6 (taking no credit for core neon radiation). This may be a result of the frequent Type III ELMs, which caused significant confinement degradation in the original H-mode experiments on ASDEX.

It is interesting that even when the plasma dropped from the CDH mode into the L-mode in

ASDEX-Upgrade, H factors of ~1.4 were sustained. This suggests a connection with the Z-mode observed on ISX-B, and also with silicon wall conditioning + neon injection results in TEXTOR reported at this conference. In these experiments TEXTOR found enhanced L-mode confinement (H ~ 1.4 – 1.7) at high density, in conjunction with feedback-controlled neon injection and high radiated power. Tore Supra also reported efficient edge radiation with < 1% neon, and an ergodic divertor. They felt that the sheared rotation associated with the ergodic divertor field pattern might have reduced transport in such a manner so as to cancel the enhanced losses associated with parallel electron thermal conduction along the stochastic fields, and edge radiation.

DIII-D, like ASDEX-Upgrade, experiences a dramatic reduction in the peak power arriving at its divertor plates with strong gas puffing (either D_2 or neon). There is also little degradation in confinement on DIII-D. Interestingly the DIII-D team was able to control the natural density rise associated with D_2 injection by pumping on the outer divertor leg. Impurities injected in the presence of this flow were well shielded from the main plasma. DIII-D has a very efficient pumping geometry which directs neutrals emitted from the strike point through a narrow aperture into the pumping plenum. It was shown that this system exhausts about 180–240 Torr–l/sec during a 210Torr–l/sec gas puff, controlling the density rise without increasing divertor heat flux. With this efficient pumping geometry the DIII-D team has also demonstrated, for the first time, significant density control of H-mode plasmas, reducing the density by ~45% compared to an unpumped plasma. (The less efficient pumping geometry on JET was reported to reduce the plasma density by only 10%.) If the DIII-D team leaves the plasma attached to the divertor pump, and turns off the gas valve, they find that the pump continues to operate, extracting particles from the wall, effectively, at a rate of 35 Torr-l/sec, while the main plasma ion content remains approximately constant. This is very promising for future long-pulse or steady-state devices, suggesting a means for wall conditioning, as well as the possibility of low-recycling across the separatrix, and therefore high-performance H-modes.

It is somewhat disturbing that in the larger devices, JET and JT-60U, successful operating regimes with good H-mode confinement and high radiated power have not yet been achieved. DIII-D reports that divertor detachment (defined as a drop in plasma pressure at the separatrix strike point) begins to occur in the range of ~1/2 of the Greenwald density limit, with little dependence on heating power – a perhaps surprising result. But the wide observation of the Greenwald density limit for edge-fuelled plasmas, almost independent of heating power, is itself a surprising result. Furthermore – and somewhat ominously for ITER – no machines seem to be able to sustain H-mode confinement at densities above about 80% of the Greenwald limit. These results together suggest that a key measure for achieving the H-mode with a detached divertor might be the H-mode threshold power at density equal to the Greenwald limit. If we imagine that gas and/or impurity puffing raises the threshold power somewhat compared to "ideal" conditions, we could have a situation of the sort shown in figure 2. Detached divertor operation is only achievable in the density range of, perhaps, 50 – 80% of the Greenwald limit; but H-mode is only achievable above some threshold power which rises with density. The size of the operating window for *H-modes with detached divertors will then scale with* $P_{exp}/P_{H,GW}$, *the experimentally employed power* scaled to the power required to access the H-mode, under ideal conditions, at the Greenwald density limit.

This is certainly an oversimplified model, but there may be a grain of truth to it[1]. If we define $P_{H,GW} \equiv 0.044\ B_T S I_{MA}/(\pi a^2)$, we find a correlation between high values of $P_{exp}/P_{H,GW}$ and successful H-mode operation with a detached divertor, as shown in Table I.

Table I. Typical Heating Powers and H-mode Threshold Powers at Greenwald Limit, for Divertor Detachment Studies Reported at this Conference

Device	$P_{th,GW}$ (MW)	P_{exp} (MW)	Detached H-Mode?
Alcator C-Mod	6	2	no
ASDEX-Upgrade	3	8	yes
DIII-D	3	8	yes
JET	6	6	no
JT-60U	13	13	no

The lower density limit for detachment may be sensitive to divertor geometry. There are indications from both JET and C-Mod that operation on a "vertical" target results in increased recycling on the highest heat-flux field lines, and so perhaps detachment at lower plasma densities, giving a wider window for H-mode operation – in the *downwards* direction. Unfortunately, n/n_{GW} scales as $(aB)\beta_N/\langle T\rangle_n$, so if β_N is fixed due to MHD constraints, and $\langle T\rangle_n$ is chosen to optimize the fusion power density, then the optimum operating density runs away above the Greenwald limit as the size and/or field of a reactor is increased. If confinement scales in the usual L/H mode manner, $nT\tau \propto (HI_pR/a)^2$, and the Greenwald limit remains inviolable, the only way to avoid this problem and still attain high fusion gain is to operate with high H, high elongation, high triangularity, and/or low q.

An important issue in divertor operation is the balance between the heat fluxes to the different divertor plates. The JT-60U team has clarified this situation with quantitative measurements showing that the heat flux in the SOL, headed for the divertor plates, is relatively in/out symmetric at high densities, independent of the direction of the ∇B drift. The heat flux to the divertor itself, however, is more symmetric when the ion ∇B drift points away from the divertor plate, because the radiated power is then relatively symmetric. With the ion ∇B drift towards the divertor, the radiated power and the heat load to the divertor plates can be as much as 2:1 asymmetric (with the lower radiation and the higher heat load on the outside). Unfortunately, operation with the ∇B drift away from the divertor increases the H-mode

1. The author expresses his gratitude to G. Vlases, J. Jacquinot, and G. Janeschitz for stimulating discussions on this topic.

threshold power by a factor of ~ 2. The DIII-D team showed the encouraging result, on the other hand, for double-null divertors that it was relatively easy to use vertical position control to balance the heat load in the top and bottom outer divertors. This balance persisted even when the heat flux was greatly reduced via strong gas puffing.

A key issue for tokamak operation is the exhaust of helium ash. TFTR results indicated that fusion-produced helium is transported across a hot-ion plasma very similarly to hydrogen or deuterium, with a diffusion coefficient of order the background thermal diffusivity. This is, of course, very encouraging news. DIII-D helium pumping experiments gave $\tau_{He}^*/\tau_E \sim 8$, with their very favorable pumping geometry. The TdeV team achieved values of this parameter of ~10. even at low density, via biased divertor operation. On ASDEX-Upgrade, with a pumping system less tightly coupled to the separatrix strike point than DIII-D's, $\tau_{He}^*/\tau_E \sim 20$ is achieved, which would not be acceptable in a reactor. Interestingly, on JT-60U it was found that fresh solid-target boronization resulted in efficient helium pumping, achieving $\tau_{He}^* < 0.5$ sec with central helium beam injection. It is important to understand that the pumped-divertor experiments reported here have all been sized such that the tokamak plasma volume divided by the divertor pumping speed is of order 2 – 4x typical values of τ_E. Thus the differences in pumping results largely reflect differences in pumping geometry. On the basis of the results presented at this conference, it may well be that we can declare the problem of helium ash exhaust solved by proper design of pumping systems.

Nonetheless, all of the mysteries of divertor physics have not been solved. Even though we can study many aspects of the edge plasma with Langmuir probes, we still do not understand the underlying turbulent transport that determines, among other things, the width of the SOL. For example results from the ADITYA tokamak showed clear signs of bursting phenomena in the SOL, indicative of non-Gaussian turbulence. Intriguing results from TEXT indicated possibilities for feedback stabilization (or destabilization!) of the edge, and may suggest an important role for parallel connection length in controlling the turbulence. A number of authors have pointed out that the SOL in a tokamak can easily be interchange unstable; perhaps these results or further experiments on the role of L_c could elucidate the underlying physics.

Heating and Current Drive

At this meeting significant progress was reported in the area of heating and current drive. Perhaps the strongest theme was the development of new uses for fast waves in the ion cyclotron frequency range. On Tore Supra, for example, it was found that direct electron heating by fast waves gave rise to significantly enhanced confinement. The Tore Supra team attributed this to an increase in shear, resulting from the enhanced bootstrap current. They point out that stored energy in the form of thermal electron pressure supports the most bootstrap current per kilojoule. On Tore Supra they have also found fast wave current drive efficiency comparable to that achieved on DIII-D. The extrapolation from these results to ITER, however, is still uncomfortably distant.

At lower frequencies the Phaedrus-T team seems to have been able to drive about 1/2 of their plasma current with Alfvén waves at $\omega \sim 0.7\Omega_{ci}$. At higher harmonics the PBX-M team, using IBW heating, discovered a way to induce an internal transport barrier using poloidal rotation driven by the IBW ponderomotive force. If this result can be replicated on larger devices it may lead to a means for density profile control in a reactor, something which could be very handy to enhance fusion output power, to adjust p' for MHD stability, or to tailor the bootstrap current profile.

TFTR demonstrated clear second harmonic tritium "minority" heating, raising T_i from 26 to 36 keV in a beam-heated plasma. This type of hot-ion mode of operation has been proposed as a "spark plug" to move reactors towards ignition, if T_i and T_e can be separated by reasonable amounts of auxiliary heating in such devices. Indeed the enhanced α power that comes from $T_i >> T_e$ may be especially helpful in ITER to help push it across the threshold to the H-mode at low density. (This is especially true if *ion* parameters are the most important for the H-mode transition.) TFTR has also demonstrated that mode conversion can be used to deposit power into the electron channel, permitting off-axis heating and potentially off-axis current drive. The single-pass absorption in this situation is much greater than for traditional FWCD.

Before we leave the ion cyclotron range of frequencies, we should mention the nice success on C-Mod, in which ICRH power of up to about 2 MW has been injected, using a dipole antenna and minority heating, with P_{rad} remaining at about 50% of the input power. This success has won the C-Mod team a small wager, and we should congratulate them. More importantly, further successes along these lines may open up the possibility of using high-Z materials in ITER and reactors, greatly enhancing divertor lifetimes against erosion due to sputtering.

The JT-60U team reported important new results on current-drive and current-profile control with LHCD. This team still holds the record for most current driven (3.6 MA) and the highest current-drive efficiency (3.5 10^{19} A/m^{2}W). Interestingly, they have explained the long-standing mystery (to me) that LHCD peaks more strongly on axis at high q than at low q – rather like Ohmic heating. They show that this follows naturally from ray-tracing calculations at low $n_{||} \sim 1.7$, when the poloidal field profile is taken into account. At high q the rays simply penetrate more deeply. The JT-60U group has also pursued current profile modification with heroic efforts such as co and counter off-axis current drive, using multi-spectral LH ($n_{||} = 2.2 + 1.3$). In this approach the high $n_{||}$ component is used to generate energetic electrons at the edge, which damp the more efficient low $n_{||}$ waves, so that they drive current near the outside of the plasma. By driving the edge current with or against the bulk plasma current they can broaden or narrow the current channel. The result is a strong time variation of l_i, but the pulse length may still be too short to see the full effect.

Tokamak Concept Optimization

The VH-mode discovered on DIII-D and reproduced on JET is an attractive reactor operating mode, since it offers the possibility of confinement 50% or more above standard H-mode scalings, suggesting that reactors could be built with $I_pR/a \sim 40$ MA, rather than ≥ 60 MA, as planned for ITER. On the basis of DIII-D and JET results it is clear that one necessary condition for this mode of operation is an ELM-free edge, which is favored by the high first-regime β limits resulting from the high edge shear associated with high triangularity. Charge-exchange spectroscopy on DIII-D indicates that helium does not accumulate any more severely in these plasmas than in Elm'y discharges. However, due to geometry constraints, helium pumping has not yet been tried in VH-mode plasmas in DIII-D. A second necessary condition for VH-modes in DIII-D was proposed at this conference. It appears that the VH-mode terminates when q(0) falls to the point where Mercier stability is violated in the core. The high-triangularity plasmas both achieve higher q(0) with NBI in DIII-D, and also have Mercier stability down to lower q(0)'s – giving a much wider operating window for VH-mode plasmas. These results suggest that with control of edge pressure gradients (*e.g.*, by active pumping to reduce edge density), and with control of q(0), it should be possible to extend VH-modes in strongly shaped plasmas to steady-state operation. The steady-state pumping requirements and current profile control requirements remain to be determined experimentally, however. In the area of pumping and recycling control, it is worth noting that a fusion reactor must necessarily have a fairly low coefficient for recycling of particles across the separatrix. If we take $\tau_{He}^* / \tau_E \sim 10$, a helium "enrichment" factor of 1/3 in the divertor gas, and equal pumping rates for D,T and He, then the entire particle inventory within the plasma must be turned over about once every 3.3 τ_E's, corresponding to a rather low net recycling coefficient if $\tau_p \sim \tau_E$, as expected.

Recent theoretical work has shown that the "ultimate" high-bootstrap configuration for steady-state operation involves a hollow j profile (since the bootstrap current peaks off axis), and so a non-monotonic q profile, with reversed shear in the core. The core region is then second-stable to ballooning modes, but the hollow current profile can be unstable to free-boundary kinks, particularly as the location of the minimum in q is moved outwards to maximize the volume of the core second-stable region. Simple theory indicates that these kinks can be stabilized by ideal walls, but become unstable to "wall modes" on a time-scale of order the resistive wall-penetration time for the mode of interest, even if the plasma is rotating. More sophisticated analyses, including ion-sound dissipation in the plasma, indicate that plasma rotation can stabilize wall-modes. Further forms of dissipation, such as resistivity, might increase the coupling of the mode to the plasma, and enhance the stability window. At this conference the DIII-D team reported that it was possible to operate well above the wall-at-infinity β limit, so long as the plasma continued to rotate. The PBX-M team also found that operation above the wall-at-infinity β limit was possible with a conducting wall. The HBT-EP team reported enhanced stability with a nearby conducting shell elements, compared to operation with these shell elements retracted. The precise rotation speed requirement seems a bit ambiguous. It is uncertain, experimentally, if the speed requirement is coupled to the Alfvén speed in the plasma (or equivalently the ion sound speed at given β), or to the wall penetration rate. Interestingly, the JFT-2M team has demonstrated the ability to rotate a plasma using AC external field perturbations.

Operation with a monotonic q profile, with $q_{min} > 2$ and very high edge q has been explored on DIII-D. It was found that in this regime the plasma was very quiet from an MHD point of view, and the central particle confinement was greatly enhanced. The TFTR team recently explored operation with $q_{min} \sim 2$, and a non-monotonic q profile. They found an impressive increase in central electron pressure, due both to increased density and temperature. Theoretically, the stability of Alfvén modes is sensitive to the q profile, so studies of α stability in DT reversed-shear plasmas will be valuable.

Interesting new results were also reported in the area of very low aspect ratio. START is now operating with a more powerful central solenoid, allowing pulse lengths of up to about 30 msec. This should permit higher quality confinement measurements in the near future. Any spherical tokamak reactor will depend on non-inductive start-up and sustainment. The HIT experimental team has now demonstrated the generation of toroidal plasma currents of up to 200kA, using helicity injection alone.

If you were to ask a stellarator enthusiast for the key area where tokamak improvement is needed, she would certainly point out that disruption control is of crucial importance to the tokamak concept. The JT-60U team presented key new results in this area, in which they showed that impurity pellet injection can be used to induce controlled disruptions which come down in plasma current relatively gently, and do not result in the production of large runaway populations. These disruptions were induced, however, in otherwise benign plasmas (a good starting point), but work is still needed to see if pellet injection can be a successful method for gently bringing down a marginally stable plasma that was already showing signs of distress. ASDEX-Upgrade and Alcator C-Mod have performed rather detailed studies of disruption effects, and particularly of "halo" currents. These results will be very important for the design of the vacuum vessel and its supports, and for the design of internal plasma-facing components, in ITER and other future devices.

So where do we stand in "advanced tokamak" performance? In some ways the most exciting new result in this area was the demonstration of a 7-second long period of $\beta_N \geq 3$ on JET. Another strong candidate for this honor is the achievement of $n_{do}T_{io}\tau_E$'s in DIII-D VH-modes that are quite competitive with the larger divertor machines, which have about 6x DIII-D's a^2B^2. But perhaps the prize should be given to the 1-sec non-inductive current drive results on JT-60U, demonstrating >100% non-inductive current drive, with 74% bootstrap current, $\beta_N = 2.9$, and $H = 2.5$ at $q_{95} \sim 5$. In short, the results presented at this meeting provide a great deal of support for optimism about the prospects for steady-state, advanced-tokamak operation.

Conclusions

The tokamak experimental results reported at this conference build on our previous understanding of these plasmas, and continue to strengthen our belief that fusion can provide humankind with a limitless, affordable, and attractive energy source. Nonetheless fusion research is in a race to develop this energy source before the world runs out of fossil fuels, or we foul our planet with the waste products of energy production. If this is a horse race, we have to ask ourselves which horse we should ride – the traditional work-horse, the inductively driven pulsed tokamak, or the wild young mustang, the so-called "steady-state advanced tokamak." Table II summarizes the advantages of each. The inductively driven core of a pulsed tokamak has no need for current-drive systems, and so minimizes expensive recirculating power. On the other hand, current drive in the core of an advanced tokamak permits continuous operation, and the resulting current-profile control that is possible may lead to higher β's and reduced transport. At the edge of the plasma traditionalists will favor the ELM'y H-mode, which has already been demonstrated to allow quasi-steady-state operation and effective helium pumping. The more adventurous horse-racer may prefer ELM-free operation, which is required so far for the highest confinement times, and eliminates the need to protect the divertor plates from repetitive pulses of energy. Some might prefer to mix and match, perhaps taking an adventurous core and a traditional edge.

In our race the reliable old work horse might be too slow. The cost of electricity could be too high, the large minimum unit size of a fusion powerplant could make it a weak competitor in some important markets, and the pulsed nature of the device could present problems of fatigue and reliability. On the other hand, the wild young mustang has not yet been fully tamed – even though we have seen good signs at this conference – and controlling that horse could be beyond our riding skills. The young mustang might well throw its riders in the ditch!

So what should we conclude? I'd say we had better continue to train both of these horses, working to overcome their weakness and to maximize their strong points.

See you at the next horse show!

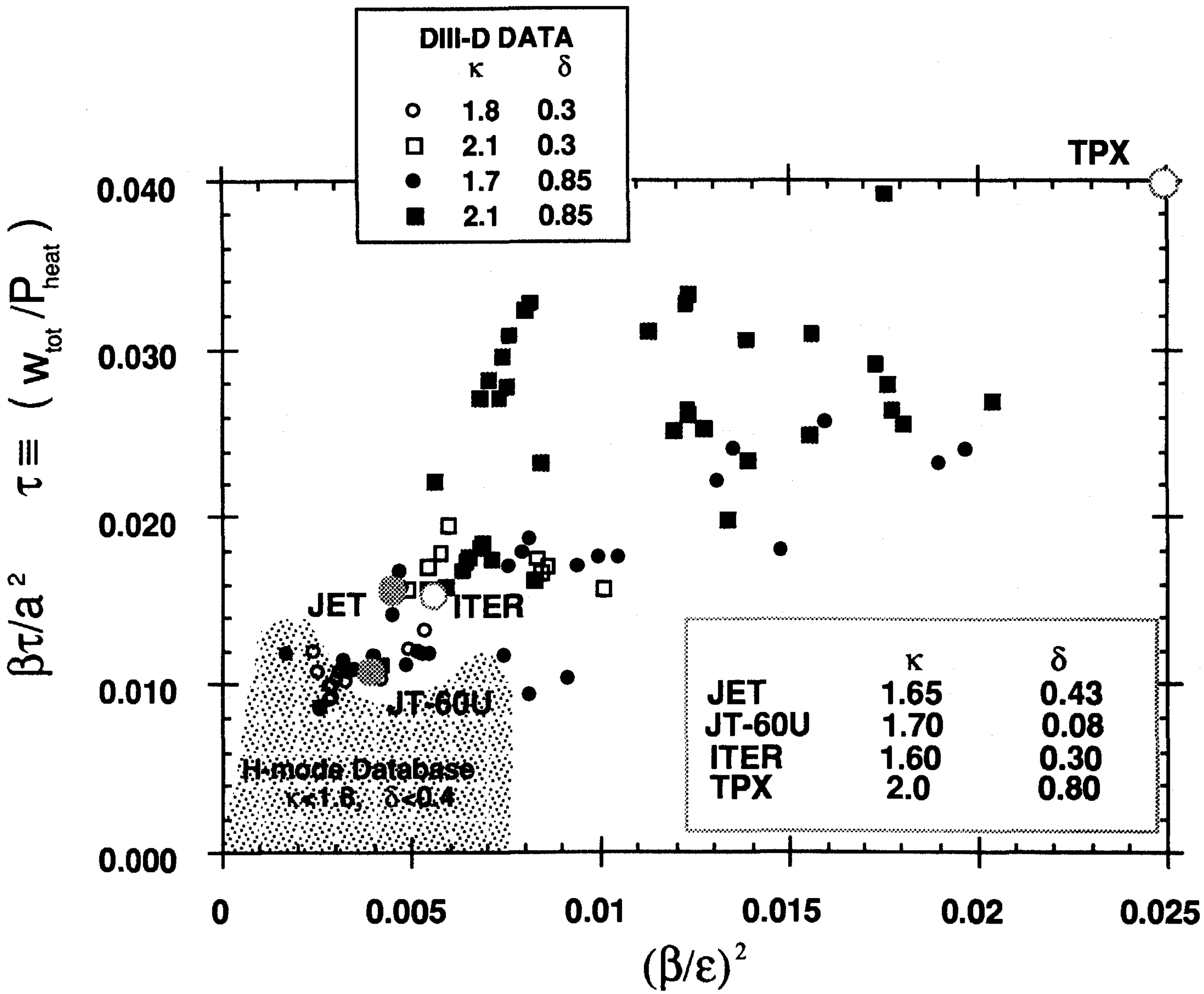
Table II. Fusion Race Horses

	Reliable Old Work Horse	**Wild Young Mustang**
Core	<u>Ohmic j(r)</u> • No need for current drive • Minimum recirculating power	<u>Non-inductive j(r)</u> • Needed for steady-state • Possibility for enhanced β & $nT\tau$
Edge	<u>ELM'y H-mode</u> • Quasi-steady-state demonstrated • Particle control demonstrated	<u>ELM-free H-mode</u> • Enhanced $nT\ \tau$ demonstrated • No heat pulses to divertor

Figure Captions

Figure 1. Highest performance JET and JT-60U data plotted against DIII-D shape-scan data, with normalized axes (see text). Also shown are results from the H-mode database, constrained for modest shaping, as well as the minimum target operating points for ITER and TPX.

Figure 2. A model for the power requirement for H-mode operation with detached divertor.



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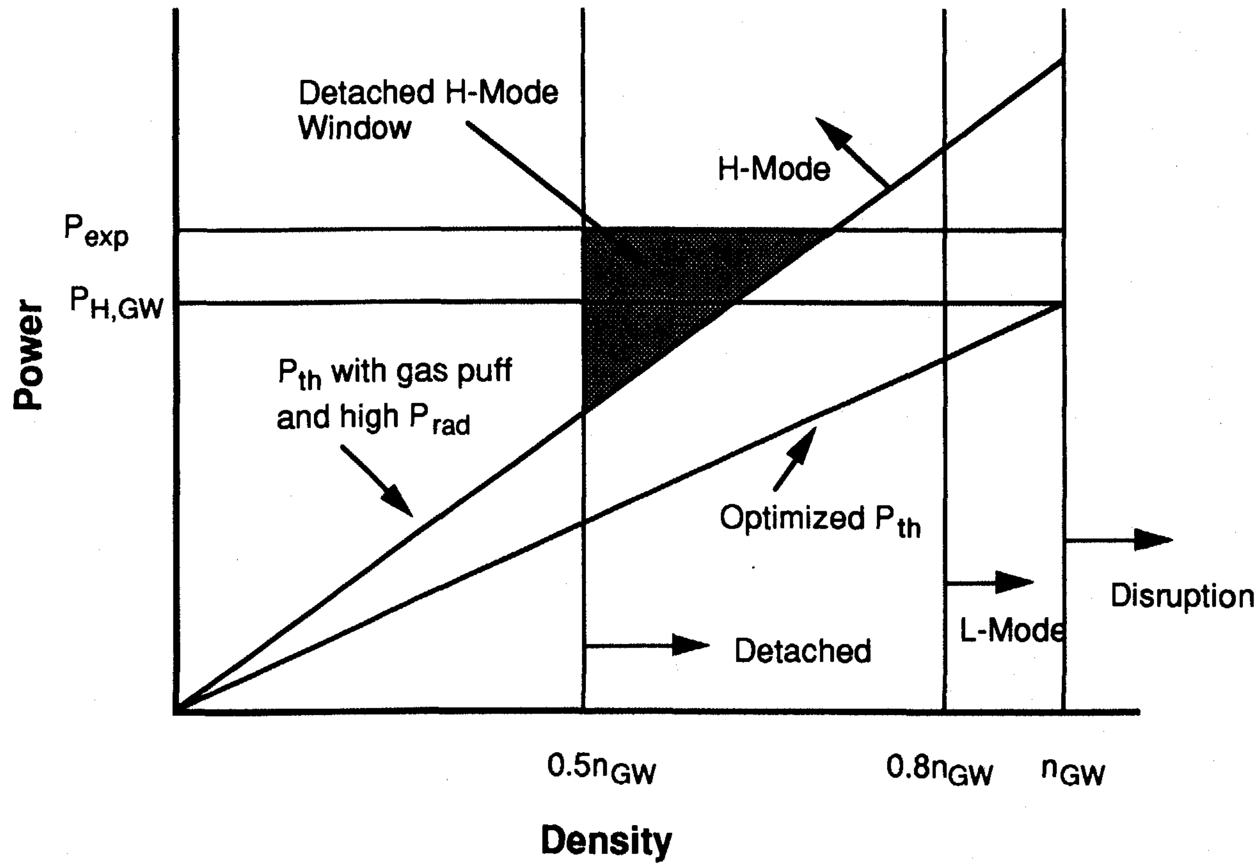

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