The cover picture shows an interior view of the Tokamak Fusion Test Reactor vacuum vessel. At the left, on the inner wall of the vessel, is the bumper limiter composed of graphite and graphite composite tiles. Ion cyclotron radiofrequency launchers are seen at the right along the mid-section of the vacuum vessel wall. By courtesy of the Princeton University Plasma Physics Laboratory.
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In four volumes

VOLUME 4

CONFERENCE SUMMARIES

INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA, 1995
FOREWORD

The 1994 International Atomic Energy Agency Conference on Plasma Physics and Controlled Nuclear Fusion Research was characterized by a multitude of excellent scientific results on virtually all aspects of controlled fusion and fusion technology. Taken together, these results lay a solid foundation for continued progress and future steps.

The conference was the 15th in a series of meetings which began in 1961 and which, since 1974, have been held on a biennial basis. The conference was organized by the IAEA in co-operation with the Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas, Asociación Euratom–CIEMAT para Fusión, Madrid, Spain, to which the IAEA wishes to express its gratitude. The conference was attended by some five hundred participants from over thirty countries and from two international organizations.

In the technical sessions, which included seven poster sessions, more than 240 papers were presented. Contributions were made on magnetic and inertial confinement systems, fusion technology, magnetic confinement theory and modelling, alternative confinement approaches and next step concepts (ITER, TPX, etc.). The traditional Artsimovich Memorial Lecture was given at the beginning of the conference.

These proceedings include all the technical papers and five conference summaries. For the second time, the summaries are being published as a separate volume before the rest of the proceedings.

The IAEA contributes to international collaboration and exchange of information in the field of plasma physics and controlled nuclear fusion research by organizing these biennial conferences as well as co-ordinated research projects, technical committee meetings, workshops, consultants meetings and advisory group meetings on relevant topics. Through these activities, the IAEA hopes to contribute significantly to the attainment of controlled fusion power as one of the world’s most important future energy resources.
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SUMMARY ON MAGNETIC CONFINEMENT EXPERIMENTS — I: TOKAMAKS

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1. 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 TFTR and the observation of unexpectedly favourable 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 ELMy operation in JT-60U, JET and DIII-D. In all three devices ELM free H modes achieved $n_{Tr}$ values about 2.5 times greater than ELMing H modes, but had not been sustained in quasi-steady state. Important progress has been made in the understanding of the physical mechanism of the H mode in DIII-D and in 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. 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. In this mode even energy dumps from ELMs did not burn through to the divertor plate. The first results from vertical plate divertors in Alcator C-Mod and JET are very promising. The C-Mod team has also shown successful ICRH 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 (FWCD) on both Tore Supra and DIII-D. At higher frequencies, lower hybrid current drive (LHCD) 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 areas of experimental tokamak physics, 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 reported which indicate 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 ways in which the current profile control tools developed in previous years are now being applied to develop new ‘advanced tokamak’ regimes on JT-60U and JET. In the sessions specifically addressing concept optimization, 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 C-Mod deepened our understanding of vertical displacement events. Perhaps the crowning achievement in this area was 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 1980s, with sawteeth in the core, high neutral recycling at the edge, L mode confinement globally and Troyon $\beta$ 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.

2. CORE PLASMA PHYSICS

A major highlight of this conference was the report that 9.2 MW of fusion power had been produced in supershots in TFTR. This constitutes an improvement of more than a factor of 5 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 limited more by gross MHD stability than by confinement. The very peaked pressure profiles in TFTR supershots lead to extremely high central fusion power density (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 about 20% higher
in DT plasmas than in equivalent DD discharges. This is attributed largely to the ion channel, where $\chi_i$ is found to scale as $\langle A_i \rangle^{-1.4}$.

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 $\alpha$ driven toroidal Alfvén eigenmodes (TAEs). The signal levels, however, were still lower than those observed to cause loss of beam ions during beam driven TAEs, and indeed no enhanced $\alpha$ loss was observed. Absolutely calibrated charge exchange recombination spectroscopy has been used to observe the slowing down $\alpha$ particles in the plasma in the energy range 80–800 keV. The measurements are in remarkably good agreement with TRANSP Monte Carlo calculations based on classical collisional thermalization. In cases with sawteeth, the central $\alpha$ density is somewhat depleted, as might be expected. After the $\alpha$ particles slow down, they are observed to transport across the plasma with $D \sim \chi_e$, which is 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 s, in order to enhance the $\alpha$ 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 ELMing H modes have been established for 20 s, with good H mode confinement. Unfortunately, the highest performance JET ELM free hot ion H modes reported at the last conference have not yet been reproduced in the new JET. Experimental results suggest that both increased recycling compared with the previous campaign and the lower triangularity of the new configuration are contributing to this problem. Interestingly, however, in both the old JET and the new, the $n_{\alpha} T_{\alpha} \tau_E$ attained in these ELM free hot ion H modes (or VH modes) is about 2.5 times greater than can be achieved in ELMing H modes. A very interesting new achievement in the JET data set is a 4 MA ELM free H mode with a density of nearly $10^{20}/m^3$, a high $n_{\alpha} T_{\alpha} \tau_E = 4 \times 10^{19} s/m^3$, $T_{\alpha} = 11$ keV and $T_{e0} = 11$ keV.

A new world record $n_{\alpha} T_{\alpha} \tau_E$ was reported in JT-60U in short ELM free phases of ‘high $\beta_p$ 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-state ELMy H modes. The JT-60U team was able to sustain the ELMy H mode in quasi-steady state for 1.5 s 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 VH 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 $\langle nT \rangle \tau_E$
about a factor of 2.5 higher than ELMy H modes. (This measure does not give credit for the high $T_i/Te$ attained in hot ion VH modes, as does $n_0T_i/Te$.) 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 $\epsilon^{-1/2}$. The parameter $\beta/\epsilon$ also equals $\beta_N f$, where $f = R_\rho/a^2 B \propto 1/q_{\text{cir}}$. The ordinate represents $<nT> \tau_B$ (where $\tau_B$ is defined conservatively as $W_{\text{tot}}/P_{\text{tot}}$) normalized to a ‘generic’ L/H mode scaling: $<nT> \tau \propto (I_p/\epsilon)^2 \propto (aB)^2 f^2$. Success in optimizing $\beta_N$ shows up along the abscissa, and success in raising $H$ along the ordinate. Along both axes, 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 $B$.  

---

$^{1}$ $B \propto \epsilon^{-1/2}$ follows from assuming that the maximum field at the toroidal coil is fixed, and that the gap between the plasma and the coil edge, on the inside in major radius, is $\sim 0.21 R_0$, e.g. a fixed 1.5 m gap with fixed $R_0 = 7$ m. Then over the range $\epsilon = 1/3-1/5$, $\epsilon B_0^2 \sim$ constant.
$\beta_N$. 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 $\beta/\epsilon$ and $\beta_T/T_e/\alpha^2$, but have been constrained out of the H mode data set 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 into confinement scaling at this conference. Both the TUMAN-3 and T-11 teams reported confinement in ohmically heated plasmas with boronized walls far above neo-Alcator scaling. At extremely 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 $\beta$ and $\nu^*$, but varying $\rho/R$. In detailed L mode scans the DIII-D team found that the electrons behaved favourably, as in gyro-Bohm scaling [$\chi \propto \chi_B(\rho/R)$], where $\chi_B \propto T/eB$, while the ions behaved less favourably than in Bohm scaling, approximately as $\chi_B(\rho/R)^{-1/2}$. (They kindly termed this particularly unattractive behaviour '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 as in gyro-Bohm scaling. 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 inwards at a speed of 150 m/s. G. 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 drop in $D_\alpha$ light, develop well before the electric fields due to toroidal rotation and ion pressure gradient build up. The timing of the poloidal rotation and of the turbulence suppression is such that it appears the rotation precedes the turbulence suppression, but this may be open to debate. The JT-60U group presented
data detailing the internal transport barrier in the ion channel that is observed in the high $\beta_p$ mode. Both rotation shear and ion temperature gradients build up first in a region inside $q = 3$, and then the steep gradient region 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 $P_{\text{tot}}/S = 0.044n_e B_T$ 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 appears also 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. F. 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, because of the high value of $\tau_E$.

3. DIVERTOR AND EDGE PHYSICS

This was the first conference where extensive results from tokamaks with pumped divertors were reported, and the results are very encouraging. The most exciting news is the achievement of completely detached H modes (CDH modes) 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. Firstly, the use of neon as a radiating impurity may be prohibited by the associated fuel depletion. In ASDEX-Upgrade $\Delta Z_{\text{eff}} = 0.6$ for $\Delta P_{\text{rad}}/P_{\text{aux}} = 35\%$ at $P_{\text{aux}}/S = 0.17$ MW/m$^2$, to be compared with ITER’s $P_a/S \geq 0.25$ MW/m$^2$. Fuel depletion increases both the $\nT\tau$ required for high fusion gain and the $\beta$ required to produce a given fusion output power. Perhaps a higher Z impurity would be more appropriate for ITER application. Secondly, 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 $\alpha$ power will be required to sustain the H mode, and therefore must be allowed to cross the
MAGNETIC CONFINEMENT EXPERIMENTS: TOKAMAKS

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 \( \sim 1.6 \) (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 the old ASDEX.

It is interesting that even when the plasma dropped from the CDH mode into the L mode in ASDEX-Upgrade, H factors of \( \sim 1.4 \) were sustained. This suggests a connection with the Z mode observed on ISX-B, and also with silicon wall conditioning plus neon injection results in TEXTOR, reported at this conference. In these experiments the TEXTOR group found enhanced L mode confinement (\( H = 1.4-1.7 \)) at high density, in conjunction with feedback controlled neon injection and high radiated power. The Tore Supra team also reported efficient edge radiation with <1% neon and an ergodic divertor. They considered that the sheared rotation associated with the ergodic divertor field pattern might have reduced transport in such a manner as to cancel the enhanced losses associated with parallel electron thermal conduction along the stochastic fields and due to 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 only a modest 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·s\(^{-1}\) during a 210 torr·L·s\(^{-1}\) 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 \( \sim 45\% \) compared with 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 extract particles from the wall effectively, at a rate of 35 torr·L·s\(^{-1}\), 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 potentially ultra-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, but this may be simply explicable. DIII-D reports that divertor detachment (defined as a drop in plasma pressure at the separatrix strike point) begins to occur in the range above about half of the Greenwald density limit, with little dependence on heating power — a perhaps surprising result. But the fact that the
Greenwald density limit for edge fuelled plasmas is widely observed, almost regardless of the 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 with 'ideal' conditions, we could have a situation of the sort shown in Fig. 2. Detached divertor operation is only achievable in the density range of, perhaps, 50–80% of the Greenwald limit; but the 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 $\frac{P_{\text{exp}}}{P_{H,GW}}$, 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.\(^2\) If we define $P_{H,GW} = 0.044B_T S_{\text{MA}}/\pi r^2$, we find a correlation between high values of $P_{\text{exp}}/P_{H,GW}$ and successful H mode operation with a detached divertor, as shown in Table I.

\(^2\) The author expresses his gratitude to G. Vlases, J. Jacquinot and G. Janeschitz for stimulating discussions on this topic.
TABLE I. TYPICAL HEATING POWERS AND H MODE THRESHOLD POWERS AT THE GREENWALD LIMIT, FOR DIVERTOR DETACHMENT STUDIES REPORTED AT THE CONFERENCE

<table>
<thead>
<tr>
<th>Device</th>
<th>$P_{H, GW}$ (MW)</th>
<th>$P_{exp}$ (MW)</th>
<th>Detached H mode?</th>
</tr>
</thead>
<tbody>
<tr>
<td>Alcator C-Mod</td>
<td>6</td>
<td>2</td>
<td>No</td>
</tr>
<tr>
<td>ASDEX-Upgrade</td>
<td>3</td>
<td>8</td>
<td>Yes</td>
</tr>
<tr>
<td>DIII-D</td>
<td>3</td>
<td>8</td>
<td>Yes</td>
</tr>
<tr>
<td>JET</td>
<td>6</td>
<td>6</td>
<td>No</td>
</tr>
<tr>
<td>JT-60U</td>
<td>13</td>
<td>13</td>
<td>No</td>
</tr>
</tbody>
</table>

The lower density limit for detachment may be sensitive to divertor geometry. There are indications from both JET and Alcator 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 downward direction. Unfortunately, $n/n_{GW}$ scales as $(aB)\beta_n/\langle T \rangle_n$, so if $\beta_n$ is fixed because of 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, $nT \tau \propto (H_L R/a)^2$, 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 $\nabla B$ drift. The heat flux to the divertor plates, however, is more symmetric when the ion $\nabla B$ drift points away from the divertor plate, because the radiated power is then relatively symmetric. With the ion $\nabla B$ 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 $\nabla B$ drift away from the divertor increases the H mode threshold power by a factor of $\sim 2$. On the other hand, the DIII-D team showed the encouraging result for double null divertors that it was relatively easy to use vertical position control to balance the heat load to the top and bottom outer divertors. (The heat load to the inner divertors is negligible.) This balance persisted even when the heat flux was greatly reduced by strong gas puffing.
A key issue for tokamak operation is the exhaust of helium ash. TFTR results indicate that fusion produced helium is transported across a hot ion plasma very similarly to hydrogen or deuterium, with a diffusion coefficient of the order of the background thermal diffusivity. This is, of course, very encouraging news. DIII-D helium pumping experiments gave $\tau_{\text{He}}^*/\tau_{\text{E}} \approx 8$, with their very favourable pumping geometry. The TdeV team achieved values of this parameter of $\sim 10$, even at low density, with biased divertor operation. On ASDEX-Upgrade, with a pumping system much less tightly coupled to the separatrix strike point than in DIII-D, $\tau_{\text{He}}^*/\tau_{\text{E}} \approx 20$ 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, $\tau_{\text{He}}^* < 0.5 \text{ s}$ being achieved 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 pumping speed is of the order of 2–4 times the typical values of $\tau_{\text{E}}$. Thus the differences in pumping efficiency largely reflect differences in pumping geometry. On the basis of the results presented at this conference, it may be that we can declare the problem of helium ash exhaust solved by proper design of pumping systems.

None the less, 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 researchers 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.

4. HEATING AND CURRENT DRIVE

At this conference 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 at mid-radius, 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 FWCD 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 half of their plasma current with Alfvén waves at $\omega \approx 0.7\Omega_{ci}$. At higher harmonics the PBX-M team, using ion Bernstein wave heating, discovered a way to induce an internal transport barrier using poloidal rotation driven by the ion Bernstein wave 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 $\alpha$ power that comes from $T_i \gg T_e$ 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 heating 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 congratulate the Alcator C-Mod team on their success in injecting ICRH power of up to about 2 MW, using a dipole antenna and minority heating, with $P_{rad}$ remaining at about 50% of the input power. Further successes along these lines may open up the possibility of using high Z materials in ITER and future power 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 the most current driven (3.6 MA) and the highest current drive efficiency ($3.5 \times 10^{19} \text{A} \cdot \text{m}^{-2} \cdot \text{W}^{-1}$). Interestingly, they have explained the long standing mystery (to the present author) 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 $n_B \approx 1.7$, 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 multispectral LH ($n_t = 2.2 + 1.3$). In this approach the high $n_t$ component is used to generate energetic electrons at the edge, which damp the more efficient low $n_t$ waves, so that these waves also 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 $\xi$, but the pulse length may still be too short to see the full effect.
5. 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 $I_rR/a \approx 40 \text{ MA}$, rather than $\geq 60 \text{ MA}$, 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 favoured by the high first regime $\beta$ 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 ELMy discharges. However, owing 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 $q(0)$ values — 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 $\tau_{He}/\tau_E \approx 10$, a helium ‘enrichment’ factor of 1/3 in the divertor gas, and equal pumping rates for D, T and He, then the entire inventory of hydrogenic isotopes within the plasma must be turned over about once every $3.3\tau_E$, corresponding to a rather low net recycling coefficient if $\tau_p \approx \tau_E$, 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 the order of 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 $\beta$ limit, so long as the plasma continued to rotate. The PBX-M team also found that operation above the wall-at-infinity $\beta$ limit
was possible with a conducting wall. The HBT-EP team reported enhanced stability with nearby conducting shell elements, compared with 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 $\beta$), 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 $q_{\text{min}} > 2$ 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 $q_{\text{min}} \approx 2$ and a non-monotonic $q$ profile. They found an impressive increase in central electron pressure, due to both increased density and increased temperature. Theoretically, the stability of Alfvén modes is sensitive to the $q$ profile, so studies of $\alpha$ 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 ms. This should permit higher quality confinement measurements in the near future. Any spherical tokamak reactor will depend on non-inductive startup and sustainment. The HIT experimental team has now demonstrated the generation of toroidal plasma currents of up to 200 kA, using helicity injection alone.

If you were to ask a stellarator enthusiast for the key area where tokamak improvement is needed, he or 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. These disruptions come down in plasma current relatively gently and do not result in the production of a large runaway population. They 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 is 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 s long period of $\beta_n \geq 3$ on JET. Another strong candidate for this honour is the achievement of $n_{e/2}T_{e/2}$ values in DIII-D VH modes that are competitive with those of the larger divertor machines, which have $a^3B^2$ values about six times greater than those of DIII-D. But perhaps the prize should be given to the 1 s 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 $q_{95} \approx 5$. In sum, the results presented at this conference provide a great deal of support for optimism about the prospects for steady state, advanced tokamak operation.
6. CONCLUSIONS

The experimental results reported at this conference build on our previous understanding of tokamak plasmas, and continue to strengthen our belief that fusion can provide humankind with a limitless, affordable and attractive energy source. None the less, fusion research is in a race to develop this new 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 workhorse, 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 simultaneously opens up the possibility of current profile control, which may lead to higher $\beta$ and reduced transport. For the edge of the plasma, traditionalists will favour the ELMy H mode, which has already been demonstrated to allow quasi-steady-state operation and effective helium pumping. The more adventurous 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. Those in the middle might prefer to mix and match, perhaps taking an adventurous core and a traditional edge.

In our race the reliable old workhorse might be too slow. The cost of electricity could be too high, the large minimum unit size of a fusion power plant 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 yet to be fully tamed — even though we have seen good signs at this conference — and controlling that young horse could prove to be beyond our riding skills. The mustang might well throw its rider in the ditch!

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So what should we conclude? I would say we had better continue to train both these horses, working to overcome their weaknesses and to maximize their strong points. Humankind cannot afford to lose this race.

ACKNOWLEDGEMENT

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SUMMARY ON MAGNETIC CONFINEMENT EXPERIMENTS — II: NON-TOKAMAK SYSTEMS

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1. INTRODUCTION

This summary presents a short review of experiments on plasma creation, heating and confinement in alternative magnetic systems such as stellarators (torsatrons/heliotrons), reversed field pinch (RFP) configurations, field reversed configurations (FRCs), mirror devices and Z pinches. About 30 papers were presented on these systems, of which 13 were from stellarators and other helical systems. We excluded from this list many theoretical papers, papers on numerical simulations and papers on reactors and technology, although they were directly connected with the specific helical properties of such systems. They will be discussed in the following summaries by P.K. Kaw and Y. Seki. It should also be mentioned that results from one of the largest stellarators, ATF, were not presented; unfortunately, experiments on ATF have been discontinued, apparently permanently. However, two new installations have come into initial operation: the $\ell = 1$ torsatron TJ-IU (Spain) and the heliac H-I (Australia). After some reconstruction, first results were delivered from the stellarator L-2(M). Seven papers on RFP investigations from RFX (Italy), MST (USA), TPE-1RM (Japan), TPE-2M (Japan), Extrap-T1 (Sweden) and SWTP (China) were presented; results from other existing RFP devices were not presented at this conference. Two papers were devoted to the study of FRCs (both from Japan), four presentations contained the latest results from mirror machines (in Japan and the Russian Federation), and one paper described initial experiments with a high current Z pinch (United Kingdom).

2. HELICAL SYSTEM EXPERIMENTS

One of the most important physical problems of stellarator research (as well as of tokamak study), and of fundamental interest for all of plasma physics, is to understand the physical nature of the (anomalous) transport in toroidal magnetic systems. This global problem can be divided into several particular and (perhaps) simpler problems:

— Is the anomalous transport in stellarators of the same nature as in tokamaks or is it different?
— Does the anomalous transport depend on device parameters (ripple amplitude, depth of the magnetic well, shear and so on) or only on plasma parameters (that is, on the magnitude and profile shape of density, temperature and ambipolar electric field)?

— Has the transport a local or a global (integral) character?

— What kinds of fluctuation are responsible for enhanced transport? Is it possible, and if so to what extent, to suppress them in order to considerably improve the confinement?

There are also some other problems which are perhaps less interesting from the physical point of view but which are very important from a practical viewpoint. They are:

— Investigation of different means of plasma parameter control (density, temperature, ambipolar electric field and plasma current profile control);
— Study of various methods of improving particle and energy confinement;
— Achievement of higher plasma parameters.

Unfortunately, up to now none of these problems has been resolved completely. However, almost all of them were discussed to some extent in the papers presented and a large amount of interesting new data was obtained. The main part of these results came from three devices — Wendelstein 7-AS (W7-AS), Heliotron-E (H-E) and the Compact Helical System (CHS) — which have come into full operation during the last few years and use different methods of auxiliary heating and plasma creation (ECRH at different frequencies and radius locations with O and/or X mode, ICRH, co and counter NBI, pellet injection, controlled gas puffing) and a large array of new diagnostics. Some new results from the L-2(M), TJ-IU and H-I devices were presented.

The following main topics were discussed during the conference:

(a) Energy and particle transport;
(b) Different methods of plasma parameter control;
(c) Edge plasma physics and phenomena of L–H-like transitions in stellarators;
(d) Electron cyclotron current drive (ECCD);
(e) ICRH.

2.1. Energy and particle transport

Many new experimental data on transport processes in stellarators were presented in papers from W7-AS (papers A1-8, A6-3) and H-E (paper A6-2). Steady state and perturbative methods are complementary tools for plasma physics investigation. Significant progress has been made in the understanding of the complex physics of heat and particle transport through comparison of the results from the two methods. ECRH is a powerful tool supporting both methods, because steady state discharges are easily maintained in long pulse operation and perturbative experiments
can be conducted in a very flexible and controlled way by amplitude modulation of the launched ECRH power. The narrow and well localized power deposition profile should guarantee a localized perturbation at an arbitrary location within the plasma which is well separated from the propagation region of the stimulated perturbation. At the same time the comparison of the electron heat diffusivity from simulated heat wave propagation with that from a steady state power balance analysis may make it possible to discriminate between different transport models. Using both these methods, a detailed study of transport processes in W7-AS was performed (papers A1-8, A6-3). The results can be summarized as follows:

(a) In contrast to tokamaks, where in general the perturbative thermal conductivity is significantly larger than the power balance one, both values of $\chi_e$ agree well in W7-AS. This result indicates that there is no significant dependence of $\chi_e$ on $T_e$, which is a basic ingredient of, for example, critical gradient or non-linear transport models. The absence of a profile resilience in W7-AS as well as in other stellarators (H-E, CHS, ATF, L-2) supports this conclusion. At the same time it was found that it is impossible to represent electron thermal conductivity as a function of local plasma parameters, e.g. electron temperature, whereas the perturbative methods and power balance clearly show the $\chi_e$ degradation with heating power ($\sim P^{-0.5}$). These last two results are rather unexpected, because the first contradicts the second. Thus the question about the local or the global character of electron heat conductivity still remains open and needs further investigation (papers A1-8, A6-3).

(b) The electron thermal conductivity is enhanced almost everywhere in the plasma column, but the degree of this enhancement is somewhat different in W7-AS and H-E (papers A1-8, A6-2).

(c) The experimental values of ion thermal conductivity and diffusion coefficient are close to their neoclassical values in the plasma centre, but in the outer plasma region they are strongly enhanced compared with the neoclassical prediction (papers A1-8, A6-2).

(d) In the transport analysis it is necessary to take into account the off-diagonal terms in the transport matrix (e.g. to explain the existence of hollow density and temperature profiles) as well as the ambipolar electric field (especially for systems with large aspect ratio) (paper A1-8).

(e) For flat or hollow density profiles the impurity level remains low, while for peaked profiles the impurity radiation strongly increases (papers A1-8, A6-2).

(f) The analysis of experimental data obtained by a coherent power modulation technique, where the perturbed electron temperature is measured by a space and time resolved ECE diagnostic, shows that a part of the ECRH power would be deposited outside the region predicted by ray tracing calculations. About 70% of the input power is absorbed between 0 and 5 cm, about 30% at a larger radius. At the same time the deposition profile width is well reproduced (papers A6-3, A1-8). This broadening of the deposition can be understood from the
influence of the superthermal tail of electrons. There are also some other experimental data which indirectly indicate the possibility of the generation of superthermal electrons during ECRH (papers A6/C-P1, A6/C-P2). Unfortunately, up to now there is no direct proof of the existence of such energetic electrons. The question about superthermal electrons and deposition profile broadening is important because it can probably resolve the above mentioned contradiction between the local and the global character of the electron thermal conductivity (see para. (a) above). If the deposition profile has some ‘pedestal’ there is no ‘pure’ propagation region for heat waves.

2.2. Plasma parameter control

Many experiments performed on different stellarators show that, in contrast to tokamaks, which have rigid profiles, in stellarators the temperature, density and ambipolar electric field profiles are rather flexible. From the theoretical viewpoint this is not surprising at all. Really, four functions (density, electron and ion temperature, and radial electric field) should satisfy the set of transport equations which contains three arbitrary (to some extent) functions: particle, electron and ion heat sources. Therefore, by varying these source functions we can modify the density, temperature and electric field profiles. (Injection of momentum in stellarators has practically no effect on profiles, owing to the strong longitudinal flow damping as a result of sufficiently large magnetic field ripples.) The large number of different methods of plasma heating (especially ECRH) and fuelling used now in experiments opens up wide possibilities to vary these sources and thereby to control the plasma parameters. Of course in the experiments we are not able to change all the profiles absolutely independently — modification of one of them leads as a rule to modification of all the others. In the papers presented, different methods for this kind of plasma parameter control were elaborated.

It was shown that in contrast to tokamaks, where parallel NBI significantly affects the ambipolar electric field, in the CHS stellarator the co and/or counter injection leads to only a rather slight change in the radial electric field. However, the additional ECRH produces clear enhancement of this field, which becomes positive at the plasma edge (owing to the electron temperature profile change). But the pellet injection for NBI heated plasmas in the H-E stellarator produces a more peaked density profile and consequently a more negative value of the ambipolar electric field, with a negative peak at half the plasma minor radius (paper A2-19).

Additional (to the central ECRH) strong heating of the plasma edge in the H-E stellarator also leads to a modification of the ambipolar electric field, to the improvement (by a factor of 4–8) of particle confinement and to a reduction in the fluctuation amplitude (paper A6/C-P1). The authors attribute this improvement to the suppression of ‘pump-out’ effects, but in our opinion the nature of this phenomenon is not yet clear.
Another method of plasma parameter control in CHS, by rotational transform change, was discussed in paper A6/C-P3. Rapid transition from one regime to another (and back) was observed by controlling the $t$ profile in deuterium and hydrogen plasmas without an obvious isotope effect of the main plasma ions on the threshold condition. This $t$ profile control experiment suggests that the presence of the $t = 1$ surface just inside the last closed flux surface (LCFS) and the internal disruption induced by interchange modes near the plasma centre are important for the transition. Although the confinement improvement at this transition is rather small, many of its characteristics are similar to those found in the L–H transition in tokamaks.

2.3. Edge plasma physics and phenomena of L–H-like transitions in stellarators

A great deal of attention was drawn to the problem of L–H transitions in stellarators and to the edge plasma physics which apparently should determine the conditions of these transitions. This problem is important not only because the H mode provides better plasma confinement but also because the investigation and understanding of the nature of such transitions could clarify the interconnection of transport processes in tokamaks and stellarators.

To understand the reason why the L–H transition was observed in W7-AS only at a value of $t$ close to 0.53, a detailed investigation of the magnetic topology was performed (paper A4-7). It was shown that plasma fluxes are diverted by the island X points. The particle diffusion coefficient in the SOL scales inversely with edge density for both the limiter and the island separatrix case. The small poloidal electric field component gives rise to convective flux contributions. At $t = 0.53$, the only $t$ value where low power, good H mode operation is possible, the space potential condition seems to be favourable for H mode development. The importance of ion temperature to the L–H transition was also discussed.

A fast L–H-like transition that has many features similar to those observed in tokamaks during L–H transitions was found in CHS (paper A6/C-P3). After the transition the density profile becomes broader, with a steep gradient at the plasma edge, the radial electric field shear increases, and electron density fluctuations near the LCFS decrease at the beginning, but gradually increase during the H phase.

The plasma flows and fluctuations near the plasma edge and their connection with transport were investigated in the H-E stellarator (heliotron) (papers A6/C-P1, A6-2). It was shown that electrostatic turbulence just inside the LCFS is governed by a certain coherent structure, while outside the LCFS the fluctuations have an incoherent structure.

Some results concerning the radial electric field, flow velocity and momentum decay times as functions of neutral pressure during biased electrode experiments in the small Interchangeable Module Stellarator (IMS) were reported (paper A6-6). The experimental data were compared with a proposed theory of L–H transition in stellarators.
2.4. ECCD

A very nice experiment on ECCD was performed on W7-AS (paper A6-3). The application of a coherent perturbation method where either the EC driven or the bootstrap current component is modulated made it possible to discriminate experimentally between these two current components, owing to their different localization in space. The experimental data on the absolute value of the driven current and its dependence on HF power density, temperature and launch angle were found to be in excellent agreement with linear theory.

2.5. ICRH experiments

The results of ICRH experiments performed in CHS (paper A6-4) should also be mentioned. In our opinion these are among the most successful ICRH experiments in a comparatively moderate size device. Usually in such devices the impurity accumulation, strong radiation losses and uncontrollable density increase lead to a 'soft collapse' and to interruption of the discharge. In CHS, after vacuum chamber wall boronization and reduction of radiation losses, a deuterium (with hydrogen minority) plasma was successfully created with $T_i = 550$ eV, $T_e = 400$ eV, $n = 2 \times 10^{13}$ cm$^{-3}$ at $P_{RF} = 600$ kW. The ICRF power was applied to the afterglow of the ECRH plasma (53.2 GHz, 200 kW) or to the NBI (40 kV, 1 MW) heated plasma. As usual the observed ion energy spectrum contained two components: the bulk and high energy tail components. Unfortunately, there is no information about the reason for discharge interruption or about what would happen if the RF pulse duration were to be increased: a soft collapse or the plasma reaching some stationary state.

2.6. Other experiments

High $\beta$ experiments were performed in CHS, with volume averaged equilibrium $\beta$ up to 2.1% (paper A2-20). This value was obtained for a high density plasma heated with two tangential neutral beams in a low magnetic field. Confinement improvement by means of turning off the gas puffing helped to produce high $\beta$. Magnetic fluctuations increased with increasing $\beta$, but this increase in fluctuations stopped when $\beta$ increased to more than 1%. Poloidal field control was applied to stop the outward shift of the plasma position with increasing pressure. These experiments are in agreement with theoretical predictions, made about 15 years ago, about the existence of the self-stabilization effect and the possibility of confinement improvement by an inward shift of the plasma column.

First results were reported on magnetic field structure measurements and initial plasma creation in two new stellarators — the $\ell = 1$ torsatron TJ-IU (paper A6-1) and the heliac H-I (paper A6/C-P4). After modification (installation of a new vacuum chamber, new power supply and new ECRH system with a power of up to 0.9 MW)
the L-2(M) stellarator came into operation (paper A6/C-P2). The main aim is to study the influence of strong ECRH on power absorption efficiency, the generation of high energy electrons (if they exist) and confinement properties.

2.7. Summary of results

The most important results on alternative magnetic systems reported at this conference can be summarized as follows:

(a) The nature of the transport in helical confinement systems remains unclear, despite long and thorough study by both theoreticians and experimentalists.

(b) Up to now it is unknown whether the anomalous transport has a local or a global character.

(c) The discovery of L-H-like transitions in stellarators and their analogy with L-H transitions in tokamaks suggest that edge transport in stellarators should have the same nature as in tokamaks. At the same time, the flexibility of density and temperature profiles in stellarators and certain other experimental data indicate that transport of the bulk plasma in stellarators should have a somewhat different nature than in tokamaks.

(d) The flexibility of plasma profiles in stellarators and their strong dependence on the form of particle and energy sources open up wide possibilities for plasma profile control and by this means for improvement of the confinement properties.

(e) So far no MHD limit of plasma density in stellarators has been found. Until now the maximum density has been determined by the heating power and by the level of radiation losses, but not by the development of MHD instabilities.

(f) There are some experimental data which indicate that ray tracing calculations do not always reflect the real HF power deposition.

3. RFP EXPERIMENTS

The RFP is an axisymmetric torus in which most of both confining magnetic fields (toroidal and poloidal) is produced by a relatively high plasma current. It is characterized by a weak toroidal magnetic field and a spontaneous reversal of this field in the outer region of the plasma. The main features of the equilibrium configuration can be rather well described by the Taylor theory (or by similar theories). But the physical mechanism which sustains such a state, resisting the diffusion of the toroidal magnetic field, still remains unclear. Moreover, in spite of hard and thorough study we have not yet achieved a more or less clear physical picture of many processes determining plasma behaviour in RFPs. It seems that RFP physics is more
complicated than tokamak or stellarator physics. Therefore time is needed to accumulate, systematize and analyse a large amount of additional experimental data. The main problems of RFP physics can be formulated as:

(a) What is the nature of the five main phenomena discovered in RFP experiments (the dynamo effect, anomalous transport, anomalous ion heating, anomalous loop voltage, and generation of superthermal electrons) and how are they linked (if indeed they are linked)?

(b) Is it possible, and if so how, to minimize all these effects (except perhaps the ion heating), but to save the reversed configuration?

There are also some practical problems similar to those for tokamaks and stellarators: to improve the confinement, to reach higher density and temperature, and to find the necessary scaling laws.

Not all but some of these problems were discussed in the papers presented.

3.1. Experimental results

The results on different operation modes in one of the largest RFP devices, RFX (which came into operation two years ago), were reported (paper C-2). It was shown that the most efficient setting-up mode in terms of volt-second consumption and reduced plasma-wall interaction is the aided reversal (when the initial toroidal flux can be comparable to or larger than the final value). Operating in this way, the confinement performance at I = 0.5 MA has been improved by a better centring of the plasma column and by reducing field errors. After the boronization of the first wall a further improvement of the poloidal $\beta$ and of the energy lifetime up to 13% and 1.4 ms has been obtained, although mode locking still affects the discharges. Preliminary operation at higher current (0.8 MA) confirms the possibility of achieving high poloidal $\beta$ (>10%).

The experiments in the MST device were devoted to the investigation of the dynamo effect, of the influence of fluctuations on transport and of the influence of current density profile on confinement (papers A6/C-P8, C-5). It was shown that several experimental data conflict with the kinetic dynamo theory and support the MHD theory: (a) the perpendicular electron temperature (~20 eV) is well below the central electron temperature (~120 eV); (b) the superthermal electron density is independent of the runaway parameter $E/E_c$; (c) the measured transport rate is such that an electron experiences many collisions in transiting from the core to the edge; (d) the MHD dynamo term has been measured to be sufficient to drive the local edge current. Probe measurements indicate that RFP particle and energy loss is governed by magnetic fluctuations inside r/a = 0.8, with energy carried out convectively by superthermal electrons (paper A6/C-P8).

The theoretical calculations of edge turbulence models for electrostatic and magnetic fluctuations and comparison with experimental observations indicate that a crucial destabilizing role is played by the long wavelength tearing modes resonant in the core.
It was shown that inductive poloidal current drive flattens the current density profile, slows the growth of $m = 1$ tearing fluctuations, suppresses their associated sawteeth, and doubles the energy and particle confinement times. These results lend strong support to the programme of fluctuation and transport suppression using current profile control in the RFP. Electrostatic or RF (e.g. lower hybrid) current drive techniques can in principle enhance and maintain profile control.

Experiments for the study of fluctuation and confinement were performed in TPE-1RM20 (paper C-3). It was found that electrostatic fluctuations with low poloidal and toroidal mode numbers play a significant role in particle transport. These fluctuations show stronger correlation with $m = 0$ than with $m = 1$ tearing mode magnetic fluctuations. This fact suggests that the particle transport is governed by edge localized electromagnetic turbulence. Very large soft X ray signals and improvement of confinement properties have been observed in the high pinch parameter region ($\theta \approx 1.8-2.0$), in accordance with the reduction of amplitude of the internally resonant $m = 1$ modes.

Characteristics of RFP plasmas with an open divertor to control plasma-wall interaction have been studied on TPE-2M (paper A6/C-P7). Both poloidal and toroidal divertor experiments have been carried out with a close fitting shell. In the poloidal divertor mode, properties of global MHD dynamo activity have been found to be similar to those without a divertor. Thermocouple measurements have shown that the heat flux is transported through the separatrix to the wall. In the quadra-null toroidal divertor mode, decreases of the loop voltage and the intensity of impurity lines have been observed without deterioration of global MHD dynamo activity. However, fast and pulsive $m = 0$ perturbations have been enhanced near the reversal surface.

Experiments in the Extrap-T1 have demonstrated an enhanced loop voltage anomaly in the high current operation regime. To explain this peculiarity a model was proposed and discussed that incorporates the coupled effect of enhanced MHD dynamo activity, associated with a narrowing of the mean resistivity profile occurring at high current, and kinetic losses due to field aligned, long mean free path electron momentum transfer to the wall (paper A6/C-P6).

Some results of RFP experiments and ultra-low-safety-factor discharges in another small device, SWIP ($R/a = 0.48/0.1$ m, $I \leq 70$ kA), were also presented (paper A6/C-P9).

3.2. Summary of results

Although the nature of many phenomena discovered in RFP experiments still remains rather obscure and needs further investigation, some progress in the understanding of RFP physics was demonstrated and many new data were obtained. The main results can be formulated as:

(a) Different methods of operation and confinement improvement were proposed and elaborated (compensation of magnetic field errors, centring of the plasma column, current profile control and so on).
(b) It was shown that some experimental results conflict with the kinetic dynamo theory and support the MHD dynamo model. However, the final choice between these two models at present cannot be made and further investigation is required.

(c) Fluctuation measurements performed on different devices indicated that in the core plasma, particle and energy transport is governed by magnetic fluctuations (tearing modes), whereas in the edge plasma, electrostatic fluctuations play a dominant role.

(d) Both poloidal and toroidal open divertor configurations have been realized. Global MHD dynamo characteristics are observed to be unchanged.

4. FRCs, Z PINCH AND MIRROR DEVICES

A conceptual design exists of a neutron lean D-3He fuelled FRC fusion reactor. To elaborate this idea, in the FIX machine a 0 pinch produced FRC plasma with $T_e + T_i \approx 400$ eV and $n \approx 6 \times 10^{21}$ m$^{-3}$ is translated into a large ($r = 0.4$ m) metal vessel (paper C-6). Though the density is decreased by two orders of magnitude, the decrease in temperature is smaller than the prediction of the adiabatic theory. In addition, when the FRC is reflected at the mirror field, plasma heating occurs owing to the thermalization of the translational kinetic energy. This re-thermalization is observed only when the translational velocity exceeds the sound velocity in the plasma, signifying the intervention of some shock mechanism. As a result of this translation, a configuration with a lifetime of about 0.5 ms is attained. In this FRC plasma, a magnetic field fluctuation localized near the separatrix is detected. High $\beta$ sheath thickness measurements show that the plasma that escapes out of the configuration seems to be flowing along the magnetic lines of force outside the separatrix.

In paper C-4 the results of two compact toroid merging experiments and the 3-D effects of magnetic reconnection were discussed. The experiments show the notable dependence of the merging speed and particle (namely ion) heating on the merging angle $\theta$. The observed effect of strong ion heating with $\theta > 90^\circ$ may be related to the anomalous ion heating observed in conventional RFPs and spheromaks but not observed in tokamak plasmas. The observed 3-D effect of reconnection can explain some of the different confinement characteristics of current carrying toroidal plasmas — FRCs, RFPs, spheromaks and tokamaks.

Only one paper was devoted to investigation of Z pinch plasmas (paper A6/C-P10). In this paper the first experimental and theoretical results from a study of a large dense Z pinch were discussed. A large pulsed power generator (2 MA at 2.4 MV for 200 ns) for Z pinch research was completed in April 1993. Its principal objective is the study of radiative collapse at a current significantly above the Pease–Braginskii current and the investigation of Z pinch plasmas close to fusion conditions. Currents of up to 800 kA at a line voltage of 1.5 MV have been discharged
in carbon or aluminium fibres of diameters in the range 7–50 μm. Discharges are characterized by a current rising to 700 kA in 200 ns. The discharge terminates abruptly after 100–200 ns, accompanied by a very hard X ray emission of 350–450 keV, depending on the fibre size and current amplitude. If CD₂ fibres are used, a neutron pulse of about 10⁹ neutrons is also observed at this time.

The mirror machines are of interest because they can provide (in principle) a reactor with advanced fuels such as D–³He and can be used as rather compact sources of neutrons. The simple axisymmetric system (with maximum B) is unstable against flute instabilities and the non-axisymmetric system (with minimum B, provided by anchors) has an enhanced radial transport. Some methods of MHD stabilization in axisymmetric systems and radial transport minimization in non-axisymmetric mirrors were developed and used.

In the GAMMA 10 tandem mirrors good radial confinement (better than the axial confinement) was obtained through the suppression of drift wave fluctuations by the naturally produced strong radial electric fields. These fields are formed when the plug and barrier potentials for axial confinement are formed. An ion temperature of about 10 keV is achieved and thermonuclear fusion neutrons are observed in ICRH plasma (paper C-1).

In the HIEI mirror device, ICRF waves are used for axial potential formation as well as for plasma production, heating and MHD stability. It was shown that when one of the limiters is positively biased, H-mode-like behaviour appears in the plasma, resulting in improvement of radial confinement by a factor of 2.5. With a combination of limiter biasing and axial potential formation by the resonant RF wave, central cell β higher than 10% has been achieved at a plasma density of ~10¹³ cm⁻³.

The possibility of producing and confining stable plasmas in fully axisymmetric mirrors with an additional semicusp was also shown in the AMBAL-M mirror device (paper A6/C-P11). Even non-axisymmetric neutral injection does not lead to MHD instability excitation.

These experiments demonstrate that it is possible to confine stable plasmas in fully axisymmetric mirror devices as well as in non-axisymmetric tandem mirror machines with sufficiently small asymmetry, owing to the stabilizing anchors.

5. CONCLUDING REMARKS

One of the characteristic features of much of the work presented at this conference was the tendency not to try to reach the highest plasma parameters by all possible means, but to undertake a deeper and more detailed investigation of different physical phenomena determining the plasma behaviour. In this way many new and interesting results were found which give us a wide field for further consideration. The thorough analysis of all these data can help to reveal more effective (and perhaps new and unexpected) ways to improve the plasma confinement and to reach plasma parameters which are as close as possible to those of fusion plasmas.
SUMMARY ON MAGNETIC CONFINEMENT THEORY

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In this paper I shall review the highlights of the magnetic confinement theory papers presented at the conference. Overall, 52 papers were presented on this topic, about 20% of the total number of papers, which shows a healthy and vigorous theory effort. Of these papers, 47 were devoted to tokamaks and only 5 to other concepts such as stellarators and pinches. This shows that the community's interest is largely directed by current experiments and the design needs for experiments in the immediate future. The tokamak papers may further be subdivided into four categories with the following distribution: ideal/non-ideal MHD studies, 21; turbulent transport studies, 18; divertor/SOL physics, 6; and RF heating and current drive, 2. The first topic is in a fairly mature state of development, the second still has major controversies, the third is in its infancy and likely to be the subject of a lot more work in coming years, and the last is largely based on theoretical concepts which are well understood.

1. IDEAL/NON-IDEAL MHD PHYSICS

In ideal MHD the overall picture is quite clear. Tokamaks must typically operate below the $\beta$ limits governed by ideal MHD instabilities such as ballooning modes and external kink modes, and we know how to calculate and optimize these limits for realistic configurations. Paper D-3 discussed the stabilizing effect of velocity shear on the ideal ballooning modes [1]. It was shown that the ballooning transformations do not lead to conventional normal modes, because the 'modes' are constantly stretched in time; the usual ansatz that $\frac{d}{dt} = -i\omega$ is therefore invalid. Using numerical methods a modified $s-\alpha$ diagram was obtained (Fig. 1) which indicates that velocity shear can produce a stability window connecting the first and second stability regimes. Paper D-5 described how realistic 3-D 'walls' (or passive conducting shells) with toroidal and poloidal gaps may be treated [2]. It was demonstrated that walls with narrow gaps are as effective as complete shells in stabilizing the low $n$ external kink modes. This result has received experimental confirmation (the HBT-EP experiment [3]). A very novel result was the complete stabilization of low $n$ kinks slowed down by resistive walls using toroidal rotation effects [4, 5]. It was demonstrated that toroidal rotation causes a coupling of unstable modes to damped sound waves (damping due to ion viscosity, ion Landau damping, etc.) and leads to a gap in the $\gamma$ versus $d_{wall}$ curve, in which the wall provides a complete stabilization against all modes,
FIG. 1. s-α diagram in the presence of velocity shear [1].

FIG. 2. Growth rate $\gamma_{\text{res}}$ and slip frequency $\Delta \omega_{\text{res}}$ versus wall radius for $\omega_{\text{res}}/\omega_A = 0.06$ [4].
ideal or resistive (Fig. 2). This result seems to be consistent with the experimental results on DIII-D [6]. An additional method using RF induced ponderomotive forces for complete stabilization of external kinks is also being explored [5].

We now turn to resistive/non-ideal MHD effects near the $q = 1$ and 2 resonant surfaces. The linear stabilization of instabilities related to sawteeth, because of the presence of energetic particles (for instance during the monster sawteeth in ICRF heating experiments), was discussed in several papers [7–9]. However, the most interesting confrontation between theory and experiment concerning sawtooth stabilization was attempted using excellent motional Stark effect (MSE) $q$ profile data from neutral beam heated TFTR experiments [9] (Fig. 3). It was concluded that the criterion for sawtooth observation must involve non-linear stabilizing physics of the ideal MHD mode. This work emphasized that linear theories alone are inadequate to explain the stabilization of sawteeth in many experiments.

The excellent $q$ profile measurements over the past few years have also led to another major puzzle related to sawtooth phenomena. Thus sawtooth relaxation at

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**FIG. 3.** Comparison of TFTR MSE data with a non-linear MHD criterion [9].
high temperature in large tokamaks is seen to take a $q < 1$ configuration to another configuration with $q < 1$. This shows that the reconnection is incomplete, in contrast to the results of Kadomtsev theory, which would predict a $q = 1$ final configuration. A new computer simulation of collisionless reconnection has indicated a possible solution to this puzzle [10]. It has been shown that collisionless reconnection, in which electron inertia provides the dominant non-ideal physics, not only proceeds rapidly, as surmised by many workers [11, 12], but also leaves the plasma with sufficient kinetic energy in flow to enable it to reorganize the magnetic configuration into a new $q < 1$ state (Fig. 4). This result is also consistent with a two stage reconnection model proposed recently by Kolesnichenko et al. [13]. In another interesting attempt to explain the $q < 1$ final configurations, a Taylor relaxation model was presented [7]. However, this model does not provide any insight regarding the dynamics of the phenomena responsible for the final configuration. It is likely that the collisionless reconnection explored in the above papers has important lessons for the study of similar phenomena in space and astrophysics.
The physics of the growth and saturation of resistive MHD $m \geq 2$ modes is in excellent shape. Several papers dealt with the locked mode instabilities in the presence of error fields, the effect of flows and resonant magnetic perturbations on $m \geq 2$ modes, etc. There were no papers dealing with disruptive instabilities.

**FIG. 4(b).** Contour plots of helical flux $\psi$ and stream function $\phi$ in collisionless reconnection: reversal due to flow [10].
Perhaps this is because the experimenters are taking the pragmatic approach: keep
away from them. If you must have one, be warned well in advance and learn how
to have a safe landing [14]! It does appear, however, that there is still considerable
physics interest in a study of these phenomena. An area of resistive MHD physics
which was conspicuous by its absence was the resistive MHD stability of negative
shear advanced configurations. These studies appear to be crucial and we are likely
to hear much more about them at the next conference.

We finally come to the topic of toroidal and global Alfvén eigenmodes
(TAE/GAE). These are Alfvén eigenmodes driven unstable by the pressure gradient
of $\alpha$ particles in DT experiments. The linear theory of these instabilities was dis­
cussed in several papers [15–18] and is in good shape. It also seems to provide good
agreement with DT experiments and simulation experiments with minority species
heating. The non-linear theories of $\alpha$ particle loss due to these modes still appear to
be in their infancy and are quite controversial. Some papers [15–17] dealt with the
energetic particle non-linearities only (trapping, stochasticity of orbits, etc.), whereas
others [18] emphasized the importance of fluid-like mode coupling effects. A detailed
simulation would involve non-linear 3-D MHD physics with kinetic effects and
would therefore be a formidable task. Hybrid and gyrofluid codes are currently under
development. In the absence of detailed codes, simple estimates of $\alpha$ particle loss due
to given amplitudes of TAE/GAE have been made. However, we need a good esti­
mate of $\alpha$ losses, and therefore a quick development of the above codes with a bench­
mark comparison with the upcoming DT experiments seem to be extremely desirable.

2. TURBULENT TRANSPORT

As observed in the introduction, turbulent transport is one of the most studied
and least understood topics in tokamak physics and is still beset with major contro­
versies. Let us summarize briefly the points of agreement. It is generally (though not
universally) agreed that electrostatic turbulence driven transport is sufficient to
explain most of the observed phenomena. It is also agreed that the edge and the core
may have separate instability drive/release mechanisms. There is also a consensus
that a tokamak plasma possesses multiple self-consistent turbulence plus velocity
shear states with different confinement properties and that it may be possible to
manipulate the plasma to go from one of these states to another. What are the major
points of controversy? There is no agreement on the most important drive mechan­
isms; all free energy sources, such as pressure gradients, curvature, impurity
gradients and atomic physics, are mentioned, and there is a lack of agreement on what
may be ignored in different regions. There is no agreement on whether the release
mechanism is linear or non-linear; if it is the latter, several decades of work generat­
ing linear intuition with regard to microinstabilities are irrelevant. There is no agree­
ment on whether the observed turbulent state is a driven, far from marginal stability
state, or whether it is a near marginal state for some as yet unknown linear or non-linear instability. In view of all the controversies, we present highlights of the results from adherents of several schools that were represented at the conference.

Computer simulations of the ion temperature gradient mode in the toroidal geometry with particle simulation codes [19] and with gyrokinetic and gyrofluid codes [20, 21] were reported. The particle simulation codes were done for relatively small plasmas and demonstrated [19] the defeat of magnetic shear due to toroidal effects, which leads to radially extended coherent structures (Fig. 5) similar to those observed in TFTR experiments [22]. It was also shown that sheared flows can lead to a breakup of coherent structures into radially localized vortices (Fig. 6). The importance of transient coherent structures and their description were also the subject of other papers [23]. The gyrofluid/gyrokinetic simulations were done for flux tubes as well as for the complete torus. Excellent agreement with TFTR experiments was claimed. It appeared that qualitative understanding of simple observations is difficult because things are buried deep in the code. An attempt to isolate important physical phenomena by switching things off one at a time could prove very rewarding. A very interesting observation from these simulations was that $\chi$ goes from $\chi_{\text{Bohm}}$ to $\chi_{\text{gyro-Bohm}}$ as $\rho_i/a \to 0$, i.e. there appeared to be a transition from one behaviour to another as $\rho_i/a$ was varied. This is a crucial observation and needs to be verified and pinned down.

The physics of L–H transition was discussed in a number of papers. Several papers [24–26] dealt with quantitative models of L–H bifurcations using ordinary differential equations, 1-D partial differential equations, etc. Interesting propagating front solutions similar to those recently observed experimentally [27] on JT-60U were reported [24]. An alternative model of L–H transition which was based entirely on the importance of diamagnetic effects on drift ballooning instabilities was also
FIG. 6. Same as Fig. 5 but with poloidal flow. Note the disappearance of radially extended structures [19].

presented [28]. Lest we take any of the models of L–H transition too seriously, an interesting new model based on atomic physics data was proposed [29]. It was shown that differing penetration depths of neutral molecules (and hence the differing source terms) in measured profiles of L and H discharges are enough to explain the differences in observed confinement. This model was more in the nature of a counterpoint illustrating the fact that we may not yet have the true story of H modes, since there is more than one way in which the observations may be explained.

The concept of non-linear stationary states initiated by non-linear instabilities was explored in several papers [30–32]. Earlier work had concentrated on pressure gradients released through resistivity effects in Ohm’s law. Now, in the collisionless regimes one is looking at gyrofluid response [30] and novel electron viscosity driven ballooning/curvature modes [31]. The latter investigation is quite interesting since it appeals to the fact that at higher temperatures, $\eta J_b$ terms in Ohm’s law should be replaced by $\mu V^2 J_b$ terms ($\mu$ being a self-consistently determined turbulent electron viscosity), a strategy which has also been invoked in collisionless tearing modes [10, 33]. Furthermore, it was shown earlier that scale invariance arguments fix the form of the turbulent transport coefficients in a manner similar to that for resistive pressure gradient driven turbulence [34]. These results are therefore likely to be generic and relatively independent of the details of the models invoked. However, it appears that detailed toroidal computer simulations are needed for making further quantitative progress.

Overall it appears that in the area of turbulent transport, more confrontation between theory and experiment (not only agreement!) is necessary to pin down the controversies and acquire some predictive ability.
FIG. 7. Evolution of magnetic islands in a heliac: (a) vacuum field, (b) rotational transform profile, (c) $\beta_0 = 1.5\%$, (d) $\beta_0 = 2.0\%$, (e) $\beta_0 = 3.8\%$, and (f) $\beta_0 = 4.4\%$ [43].
3. DIVERTOR/SOL MODELLING

Recent experiments on pumped gas target divertors have uncovered the interesting regime where the plasma detaches itself from the divertor plates. At such fronts, the plasma momentum can be shown to be taken up by charge exchange collisions (and $E \times B$ convection effects) and the incoming energy by electron excited radiative losses. Simple models describing the physics of detached ionization fronts have been constructed [35] and detailed simulations are also in progress [36]. Experiments on limiter and divertor plate biasing [37, 38] have shown SOL broadening, symmetrization of particle and heat fluxes between inner and outer plates, etc. It is therefore important to investigate the effect of biasing on the linear drives in the SOL; such an investigation was reported [23]. Finally, in an interesting simulation, the importance of chemical sputtering from divertor plates in the modelling of high density gas divertors was also demonstrated [39]. Overall the divertor/SOL modelling area is still in its infancy and deserves a lot more attention.

4. RF AND ALTERNATIVE SYSTEMS

With respect to RF current drive in tokamaks, interesting synergetic effects of the combined use of ICRF and LHCD were discussed and explained in terms of an efficient mode conversion to ion Bernstein waves [40]. The use of $\alpha$ particle pressure gradient driven lower hybrid waves for current generation is an exciting new idea needing further exploration [41], especially in view of the proposed experiments on TFTR. Theories of ICRF heating of heliotrons were presented [42] and concentrated on detailed consideration of energetic particle orbits, self-consistent electric fields, collisional effects, etc. Overall the physics of RF induced heating and current drive in fusion grade plasma seems well understood and now it is primarily a matter of finding applications to new ideas and new situations.

In the physics of 3-D equilibria (of relevance to alternative system configurations), interesting computer studies on finite $\beta$ effects demonstrating the possibility of island formation and self-healing were discussed [43, 44] (Fig. 7). Interesting studies on kinetic effects in the physics of reversed field pinch phenomena were also reported for the first time [45].

5. CONCLUSION

It is noteworthy that in earlier IAEA conferences the mood in tokamak research was one of accepting what one got, namely a monotonically rising $q \approx 1-3$ plasma, sawtoothing, $\beta < \beta_{\text{Troyon}}$ everywhere, an L mode confinement deteriorating with auxiliary power input (perhaps because of profile consistency effects), etc. At this conference, there was a buoyant new mood for bold initiatives arising because of our
ability to control fusion grade plasmas and manipulate them into giving better performance (H modes, VH modes, $\xi$ shape control, $E(r)$, $V_{\theta,\phi}$, $p(r)$ control, negative shear possibilities, etc.), which shows the way to the new 'promised land' of advanced configurations. Here we may use a controlled non-monotonic q profile to give us a discharge with no sawteeth, $\beta_{\text{core}}$ in the second stability regime, high bootstrap current, confinement in the H/VH mode, etc. The contributions of theory in the form of generating an overall (primitive) understanding of what is going on, some predictive ability which helps in designing new experiments and as a source of new ideas, are very significant in the development of the present mood. It is hoped that magnetic confinement theory will continue to prosper and that at the next IAEA conference we will see many more contributions along these lines.

REFERENCES

[31] ITOH, K., et al., IAEA-CN-60/D-17, these Proceedings, Vol. 3;
[35] GHENDRIH, P., CAPES, H., IAEA-CN-60/D-P3, these Proceedings, Vol. 3;
[38] TAMAI, H., et al., IAEA-CN-60/A1-7, ibid.
SUMMARY ON INERTIAL CONFINEMENT FUSION

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1. HIGHLIGHTS

1.1. The US National Ignition Facility

Inertial confinement fusion (ICF) aims at controlled thermonuclear fusion in small capsules containing deuterium–tritium (DT) fuel. The first laser facility designed to actually ignite and burn a capsule is now planned in the United States of America under the name of the National Ignition Facility (NIF). As M. Sluyter from the US Department of Energy told the conference, conceptual design and project planning for the NIF have been completed and the decision process is in progress. The NIF driver is a neodymium glass laser and is expected to take approximately a decade for design and construction, with an associated cost of the order of US $1000 million. A large number of ICF contributions to this conference refer to the NIF; more detailed information on the NIF laser and the NIF target design will be given below. For Europe, it is of particular interest that France has plans to build a machine similar to the NIF in close co-operation with Lawrence Livermore National Laboratory (LLNL), as announced by the laser team from the Centre d'études de Limeil-Valenton (CEL-V).

1.2. Demonstration of diagnostics for hot spot formation

Ignition is expected to occur after target implosion in a tiny DT fuel volume (the hot spot) about 100 μm in size and with a lifetime of 100–200 ps. Hot spot formation inside colder and more compressed fuel is a key problem for ICF. New diagnostics with unprecedented space and time resolution have been developed by the group at the Institute of Laser Engineering (ILE), Osaka, Japan, and represented a highlight of the conference. The new technique produces a sequence of 2-D images in time intervals of 10 ps and with 15 μm spatial resolution. This makes it possible for the first time to follow hot spot dynamics in great detail.

1.3. Declassification of hohlraum target design

A highlight of a different nature concerned hohlraum targets. Thanks to declassification steps enacted by the US Government in 1993, it was at this IAEA conference that the design of such targets was presented in detail for the first time.
This refers mainly to work in the USA, but hitherto classified work in France was also reported. These contributions included a variety of results on hohlraum experiments in the past and discussions of present designs as well as critical issues still to be solved. First ignition and burn is now expected to be demonstrated with hohlraum targets. Open communication on these central issues of ICF is a very important point. It is creating new options for collaboration and will stimulate ICF research worldwide. It was also clearly stated that the declassification step does not violate any non-proliferation rules, a concern which had seriously hampered the development of inertial fusion on an international basis, in particular when compared with magnetic fusion.

2. FUSION TARGETS

2.1. Hohlraum target design for the NIF

Hohlraum targets rely on conversion of beam energy into soft X rays in small cavities and indirect drive of the fusion capsule by these X rays inside the cavity. The rationale for indirect drive is to achieve sufficiently symmetric target implosions. With the veils of secrecy lifted, these targets were broadly discussed by many groups. It turned out that the conceptual ideas as well as scaling properties and basic parameter values concerning hohlraum targets had been more or less known in the open literature before. However, what matters is the quantitative detail in which formerly classified material is now available and can be quoted. Also, the possibility for direct discussion of the results with the authors is re-establishing the rules of normal scientific communication in this field.

The baseline target to achieve ignition at the NIF was presented by J. Lindl. It consists of a cylindrical hohlraum 9.5 mm long and 5.5 mm in diameter. It contains a spherical fusion capsule of 1.11 mm radius which consists of a doped plastic ablator, a cryogenic DT layer and DT gas in the centre. All sizes, densities and material compositions are given, as well as important details such as the surface finish of the capsule and the location of the laser interaction spots on the hohlraum wall. The temporal shape of the 15 ns long pulse starts with a foot at 10 TW power and then rises to 410 TW. In order to achieve the required X ray uniformity of 1–2% on the capsule, hohlraum cases with an area 15–25 times larger than the initial capsule area are considered; this limits hohlraum coupling efficiencies to 10–15%. The hope is to improve this coupling efficiency eventually to 20–25% through the use of optimal driver and hohlraum geometries.

The NIF target can be ignited with 1.3 MJ laser pulses and has a gain of 10, according to calculations performed at LLNL and independently at Los Alamos National Laboratory (LANL). These calculations are based on theoretical models which have been successfully tested in experiments with smaller scale targets. It is interesting to note that there exists no simple global scaling for hohlraum targets.
Whereas most experiments with the Nova laser at LLNL were carried out with vacuum hohlraums driven by 1–4 ns pulses, the larger NIF hohlraums driven by 15 ns pulses need to be filled with some buffer gas to slow down wall ablation, which otherwise would change the hohlraum geometry and radiation symmetrization in an unacceptable way. Of course, this introduces hydrodynamic coupling between the region of laser deposition and the capsule and may have a negative effect on implosion symmetry. As M. Cray from LANL told the conference, the LANL group has developed integrated modelling, combining all relevant physics to model hohlraum performance down to capsule ignition in a single simulation run. The calculated results show that the gas filled hohlraums work successfully; however, experiments for probing particular aspects of the concept are still in progress.

2.2. Hohlraum experiments

A basic hohlraum experiment is to measure the radiation temperature through a diagnostic hole as a function of the beam power injected into the cavity. Results obtained in the past at LLNL and at CEL-V were reported for the first time and are in good agreement with results already published by other groups. The results obtained at LLNL with 30 TW pulses of 0.35 \( \mu \)m laser light from the Nova laser showed temperatures of up to 280 eV. According to the Stefan–Boltzmann law, such a temperature corresponds to \( 6 \times 10^{14} \) W/cm\(^2\) of radiation flux in the hohlraum, the flux level needed for ICF targets. Scaling properties are understood in terms of a model based on power balance.

A second type of experiment studies radiation symmetrization in the hohlraum by observing initially spherical capsules imploded deliberately to pancake or sausage shapes depending on the geometry of laser deposition. Good agreement between observations and corresponding simulations is found. Experiments at CEL-V using hohlraum irradiation with just the two beams of the Phebus laser indicated that an RMS X ray non-uniformity on the central capsule of 15% could be obtained; in these experiments the beams enter axially from opposite sides, and conversion takes place on two discs located on the axis and screening the inner capsule from direct illumination.

Hohlraum implosion experiments were also reported from the group at ILE and from the group at the Institute of Nuclear Physics and Chemistry in Sichuan, China. In the ILE experiments performed with the Gekko XII laser, the X ray driven capsule implosions were diagnosed with a new high resolution imaging technique which revealed time dependent low-mode asymmetries of the imploded core, even though time integrated imaging showed fairly spherical compression. The new diagnostic and results obtained with it are described further below.

The hohlraum concept implies high losses of energy into the cavity wall. Details of X ray diffusion into walls of different materials determine the transfer efficiency, which is the net fraction of X ray energy finally absorbed by the capsule. Whereas high Z elements (e.g. Au, U) with high opacity are used for the cavity wall, low Z
materials (e.g. CH, Be) are favoured for the capsule ablator. What matters is the albedo factor, i.e. the fraction of radiant energy re-emitted by the heated wall; it should be high for the casing and low for the capsule. K. Eidmann from the Max-Planck-Institut für Quantenoptik (MPQ), Garching, Germany, reported on joint MPQ-ILE experiments in which the re-emission of different materials was measured in cavities heated to 125 eV with Gekko XII; the flux re-emitted from gold was found to be 4-5 times larger than that from carbon. In the same campaign, supersonic radiation waves in 80 mg/cm$^3$ high Z doped plastic foam were investigated and also very uniform X ray driven shock waves with pressures of between 3 and 20 Mbar (between 0.3 and 2 TPa). Observation of shock break-out from wedge shaped plates placed in the hohlraum wall (‘witness plates’) is now used as a very efficient hohlraum temperature diagnostic by a number of groups.

2.3. Direct drive implosions

The direct drive approach to ICF could provide higher gain at lower incident energy than the indirect drive approach, provided that sufficient uniformity can be obtained. As S. Nakai from ILE pointed out, laser energies of 300–500 kJ are required for ignition and modest burn, according to simulations at ILE based on a 2-D implosion model and assuming 30 nm surface finish of the target as well as 1% RMS irradiation non-uniformity. In the design of the NIF target chamber, an option for direct drive with the corresponding beamline geometry for near spherical illumination is planned for the case that the techniques of beam smoothing can be sufficiently improved and the problems related to Rayleigh–Taylor instability (RTI) can be overcome. The Omega Upgrade laser now under construction at the Laboratory for Laser Energetics (LLE), Rochester, New York, is designed to address these problems in detail.

Numerical work on high gain direct drive targets was reported by S. Bodner from the Naval Research Laboratory (NRL), Washington, D.C. A carefully time shaped laser pulse of 3.36 MJ energy drives a simple spherical target (a layer of DT wetted foam, a layer of cryogenic DT, and DT gas in the centre) to a gain of greater than 200. The coupling efficiency from beam to fuel is 9.6%, taking a moderate laser intensity of $3 \times 10^{14}$ W/cm$^2$. Surface perturbations (due to target fabrication imperfections or laser imprinting) of 10–30 nm or less are required for imploding such a target successfully. The high gain is related to the high uniformity, because it allows the target shell to be imploded on a lower adiabat, which leads to high compression and efficient burn.

2.4. Ablative RTI

Ablative drive of material layers is inherently hydrodynamically unstable, because ablating low density plasma accelerates non-ablated high density material. Such a configuration is subject to RTI. This means that small surface perturbations
will grow and may eventually destroy the accelerated layer. Fortunately, extensive experimental and theoretical work during the last ten years has shown that ablation and heat conduction reduce the classical RTI growth rate by effectively a factor of 2.

New studies on RTI were presented by several groups, including ILE experiments with premodulated planar targets which were ablatively driven and diagnosed by X ray backlighting. The RTI is seen to develop up to the bubble and spike regime, in close agreement with 2-D numerical simulations. Experimental observation of instability growth was also reported by L. Dhareshwar from the Bhabha Atomic Research Centre, Bombay. Concerning RTI calculations, S. Atzeni from the Centro Ricerche Energia, Frascati, Italy, showed detailed results obtained with his DUED code, verifying the Takabe formula for ablative RTI growth and following multimode RTI dynamics into the regimes of non-linear growth and turbulent mixing.

2.5. Laser imprinting

During recent years, much attention has been given to initial perturbations seeding RTI growth. Of particular concern are perturbations imprinted by the laser on the target surface at early time when ablation is just setting in and the density gradients are still high. Simulations at NRL showed that the effect of laser imprinting is equivalent to imperfections in surface finish and can be described by equivalent amplitudes.

O. Willi from the Imperial College of Science, Technology and Medicine, London, presented a special technique to counteract laser imprinting. He covers the main shell with low density (a few milligrams per cubic centimetre) foam topped with a very thin high Z layer. Incident laser light turns the high Z material into a plasma which is strongly radiating and uniformly ionizes the foam by means of a supersonic radiation wave. As a result, a uniform buffer plasma forms almost instantaneously and prevents imprinting. In a sense, this is a hybrid scheme combining elements of indirect and direct drive. Though the capsule is driven essentially by direct illumination, symmetrization by X ray transport plays an important role during an initial period. First experiments appear to corroborate the feasibility of this concept.

A similar scheme was discussed by the group from the Instituto de Fusión Nuclear (DENIM), Madrid, with a gold burn-through foil placed at a distance from the main target layer such that a short X ray pulse shines on the target before the main laser pulse arrives. The group has performed first experiments with the Phebus laser at CEL-V to explore the scheme. The group at NRL plans to apply another variant of this idea: a startup single joule X ray flash from a thulium source is expected to reduce the laser imprint to less than an equivalent of 10 nm perturbation. The technique will be tested in Nike experiments.
2.6. Hot spot formation and mixing

The formation of an igniting DT volume in the centre of the implosion (a hot spot) is possibly the most critical point in ICF research and was discussed in several contributions to the conference. One should recall that, in the mainstream approach to high gain ICF, only a small fraction of the fuel has to form the hot spot and the bulk of the fuel is compressed on a lower adiabat to much higher density. Ignition has to occur in the hot spot and then propagate through the fuel reservoir. The problem is that the formation of the hot spot is again RTI unstable. The imploding dense fuel has the tendency to mix with the lower density hot spot during stagnation, and this may degrade ignition. Also for volume ignition, when all the fuel has to reach ignition conditions, mixing with surrounding pusher material (typically of higher Z) is a great problem. Small wavelength perturbations tend to grow to the turbulent regime under these conditions and form a kind of boiling layer which quickly grows at the unstable interface. The neutron yield is very sensitive to mixing, and indeed the yields measured so far in (non-ignition) experiments typically lag behind theoretical 1-D predictions.

H. Takabe from ILE presented 1-D implosion calculations including turbulent mixing described as a diffusion process. In these calculations the diffusion coefficient (mixing length) and the level of turbulence at an initial time enter as free parameters. Experimental neutron yields can be described in this model with reasonable values for these parameters. Depending on the fuel convergence ratio, actual yields are smaller by an order of magnitude or more than in naive 1-D predictions which do not take mixing into account.

2.7. Hot spot implosion experiments at LLNL

J. Kilkenny presented results from hot spot implosion experiments at LLNL. These were X ray driven implosions of glass microballoons filled with deuterium gas at 25–200 atm (2.5–20 MPa). Radial convergence of up to 24 and averaged fuel density of up to $19 \pm 1.5$ g/cm$^3$ were obtained. Primary neutron yields for these implosions were in agreement with simulations using the Haan mixing model. The calculated results imply 30–40% of radial mixing. Measured temperatures of fuel ions were $0.9 \pm 0.4$ keV, corresponding to a maximum fuel pressure of 16 Gbar (1.6 PPa). The burn duration for the 100 atm capsules was measured to be $50 \pm 15$ ps, giving measured values of $n_r = 1.9 \times 10^{14}$ s/cm$^3$.

2.8. Time resolved hot spot dynamics at ILE

In order to make further progress on the way to ignition, time and space resolved measurements of hot spot formation are of key importance. New techniques of X ray imaging have been developed at ILE and were described by H. Nishimura. Firstly, a monochromatic X ray imaging method was discussed which is based on
toroidally bent crystals developed by the group of E. Förster at the Max-Planck-
Arbeitsgruppe Röntgenoptik, Friedrich-Schiller-Universität, Jena, Germany. Time
integrated 2-D images of an imploded plasma taken in the light of two selected argon
lines were shown as an example. They allow the temperature distribution to be
mapped out in two dimensions.

The second diagnostic novelty from Japan was a fast X ray framing method.
The basic idea is to streak a somewhat oblique array of pinhole images over a camera
slit. In this way multiframing X ray images of a compressed core can be taken with
a frame interval of 10 ps (or less). Computer image processing allows a full 2-D
image to be obtained for each time. The reported spatial resolution was 15 μm. A
key element of this diagnostic (called MIXS) is the fabrication of very accurate
pinhole arrays and a quasi-distortion-free streak camera. A series of 25 frames taken
over an implosion stagnation interval of 200 ps showed rapid variations in the shape
of the hot spot. Such observations had previously not been possible; conventional
X ray framing cameras currently have time resolutions of down to 80 ps and merely
average over this fine structure in spot dynamics.

T. Yamanaka, also from ILE, reported on implosions of cryogenic liquid
deuterium and deuterium gas targets. He compared neutron yields with 1-D simu-
lations and pointed out that the collision of the first reflected shock with the pusher
is of special importance in the analysis of these implosions. The measured neutron
yields correspond closely to the 1-D neutron yields calculated up to this collision
time. The conjecture is that fuel–pusher mixing by the Richtmyer–Meshkov insta-
bility terminates neutron production. Time resolved X ray imaging using MIXS
corroborates this conjecture, showing that X ray emission terminates shortly after
first shock collision, about 150 ps before maximum compression is obtained in the
1-D simulation. The Richtmyer–Meshkov instability is a variant of RTI occurring at
interfaces when a shock passes. These findings appear to be highly relevant for
further understanding and improving the ignition process.

2.9. Laser–plasma interaction

Contributions on the problem of laser–plasma interaction were mainly related
to the study of parametric instabilities in hohlraum targets. In particular, ignition
scale targets with large dimensions and long time-scales involve underdense plasmas
of a scale length of several millimetres through which intense light has to be trans-
ported. In these situations, stimulated Raman scattering (SRS), stimulated Brillouin
scattering (SBS) and beam filamentation are of particular concern. For example, SRS
produces unwanted fuel preheat by superthermal electrons, whereas SBS backscatters
light and thereby reduces beam–target coupling.

Plasma conditions expected in NIF targets have been reproduced in Nova
experiments. B.J. MacGowan from LLNL reported that SBS levels, greater than
30% in some experiments, were reduced to 0.3% by beam smoothing and by enhanc-
ing ion sound wave damping through the choice of appropriate material. Similar
results have been obtained by the LANL group. It is expected that the laser-plasma instabilities relevant to NIF targets can be successfully controlled.

2.10. Light ion beam targets

Ion beams represent an important alternative to lasers for driving fusion targets. The differences are mainly related to the drivers and will be discussed further below. Targets for laser and ion beams are basically similar; however, there are differences in the way energy is deposited. Whereas laser light heats material at the surface, ions penetrate and heat the volume. Light ions (p, Li, C) and heavy ions (e.g. Xe, Bi) are very similar in respect of deposition physics, and target experiments now becoming possible with light ions are highly relevant for heavy ion fusion.

M.K. Matzen from Sandia National Laboratories (SNL), Albuquerque, New Mexico, reported on the first lithium driven hohlraum experiments at 1400 TW/g with the Particle Beam Fusion Accelerator (PBFA II). Lithium ions of 9 MeV energy in a 17 ns power pulse were radially focused to intensities of $1.4 \pm 0.4$ TW/cm$^2$ and produced hohlraum temperatures of $58 \pm 6$ eV in 5 mg/cm$^3$ foams. Though this is considerably less than what is readily produced with lasers, it signals remarkable progress in the development of light ion beams, and SNL expects to achieve 100 eV with 5 TW/cm$^2$ lithium beams in a next step by decreasing beam divergence and energy losses in the diodes.

According to the paper from SNL, light ion beams of 50 TW/cm$^2$ are required for driving ICF targets. Target design at SNL considers fusion capsules embedded within spherical foam filled hohlraums. The foam stops the ions and becomes transparent to X rays when heated. Since pulse shaping of light ion beams is difficult to achieve, a technique of internal pulse shaping is developed by means of special capsule design making use of an additional outer ablator layer of enhanced opacity.

2.11. Heavy ion beam target design

The target design just described for light ions has also been considered for heavy ion beams. New work in this direction was reported by S. Atzeni with special emphasis on pulse shaping and implosion stability. A dilemma for heavy ion target designers is that while high energy ions, e.g. bismuth ions of 10 GeV and more, are favourable for accelerator driver design, such beams have ranges of some 100 mg/cm$^2$ and therefore require a large amount of material for beam stopping. This reduces coupling efficiency and gain. With a 12.5 MJ pulse of 8 GeV bismuth ions, Atzeni finds a best gain of 30 for this design.

The LLNL design for heavy ion hohlraum targets was openly discussed for the first time by D. Ho. It consists of a cylindrical hohlraum with two localized converters placed axially on both sides of the casing. The deposited beam energy is converted into X rays with an efficiency of 50–90%, depending on converter size and ion energy. Lead shields screen the fusion capsule from line of sight radiation from
the converters, and two additional ring shaped shields control the $l = 4$ asymmetry of capsule illumination. A radiation coupling efficiency from converter to capsule of 25% is obtained and a gain of 90 for an optimized case. A very similar target design was presented in another contribution; it was devised by J. Ramirez, from the Universidad Politécnica de Madrid, and the author of this summary using the new code MULTI-2D developed by R. Ramis, also from the Universidad Politécnica de Madrid.

3. DRIVER TECHNOLOGIES

3.1. Glass lasers

Among the different types of high power laser for ICF research, glass lasers are the most advanced, and research scale facilities are now available in many countries. The major facilities suitable for target implosion experiments are located in the USA, in Japan and in France. The Nova laser at LLNL is currently the largest laser and delivers up to 40 kJ of 351 nm light on targets. The beams are arranged in cones incident from opposite sides and adequate for hohlraum targets. The Phebus laser at CEL-V provides 8 kJ in two opposing beams which are very similar to the Nova beams. The Gekko XII laser at ILE has 12 beams and provides 10 kJ of frequency tripled (351 nm) light for both direct and indirect drive experiments. The Omega laser at LLE is now being upgraded to 30 kJ of frequency tripled light in 60 beams for direct drive experiments.

Besides these leading laser facilities devoted to ICF research, there exist a number of smaller glass lasers used for a large variety of basic physics and fusion related experiments. They deal with problems of laser-plasma instabilities, laser driven hydrodynamics, X ray generation and radiation transport, as well as studies on shock waves and high energy density in matter. These activities are significantly contributing to the understanding of the physics underlying inertial fusion.

The research on glass lasers has concentrated on the improvement of beam quality. Uniform irradiation is not only crucial for direct drive experiments but is also important for indirect drive, owing to the sensitivity of laser-plasma instabilities to non-uniform intensity profiles of the beams. Methods of beam smoothing are being improved continuously. For example, the ILE group is applying combinations of random phase plates (RPPs), partially coherent light and smoothing by spectral dispersion (SSD). By these means the RMS non-uniformity in typical laser spots could be reduced from 36% (non-smoothed) to about 4%. It was stated that further reduction to less than 1% appears feasible even for the 12 beam Gekko system by making use of envelope control of the focused beam pattern by means of optimized RPPs and aspherical multilens arrays. Another area of improvement concerns power balance control to accomplish precision operation of multibeam laser systems. A campaign in this direction at LLNL under the name of Precision Nova has led to
improved target shots. Efforts at ILE are still in progress with the goal of achieving power balance at a level of a few per cent.

There are three major new facilities currently under development. The group at ILE reported on the design of a high power glass laser system called Gekko XII Upgrade or Kongoh. Relevant optical technologies for this laser have been developed. The system is to provide 300 kJ of frequency tripled light for direct drive ICF. Already under construction is the Omega Upgrade laser at LLE. It will produce 30 kJ of 351 nm light in a 40 TW shaped pulse and is intended to explore the pre-ignition regime using direct drive. With 60 beams, an illumination uniformity of 1–2% is expected using SSD.

The most ambitious and far reaching project is the NIF, which is intended to ignite and burn a target for the first time by means of indirect drive. To achieve this goal, the NIF laser will deliver about 1.3 MJ of 351 nm light in 192 beams to the target at a power of 400 TW. Conceptual design and project planning have been completed and the decision process is under way. A scientific prototype glass laser called Beamlet closely represents the performance of one NIF beamlet. It has demonstrated the four pass NIF laser structure and is scheduled to reach its energy milestone of 5 kJ at 351 nm wavelength by the end of 1994. The group at CEL-V announced that they plan to build a megajoule laser in France similar to the NIF in close co-operation with LLNL.

3.2. KrF lasers

Krypton fluoride lasers are attractive candidates for ICF because of their intrinsic short wavelength (250 nm), broad bandwidth (>0.2 nm), pulse shaping capability, high efficiency (>10%) and capability for pulse repetition. Currently the largest KrF programme is being pursued at the Electrotechnical Laboratory in Tsukuba, Japan. The existing ASHURA laser with 660 J in six beams is being upgraded to Super-ASHURA, which aims at 8 kJ on target with 12 spatially smoothed and temporally tailored pulses for ICF target experiments. The design of the main amplifier (gas volume: 60 cm diameter × 2 m length) will give high energy extraction efficiency (>10% of deposited e beam energy), high output intensity (>10 MW/cm²) and high stage gain (>50) while keeping amplified spontaneous emission at a negligible level. Completion is expected in 1995.

The other KrF system reported on at the conference is the Nike laser now under construction and almost completed at NRL. It is designed to produce 3 kJ in 44 angularly multiplexed beams on planar targets in a 4 ns shaped pulse. At the time of the conference the first phase of Nike had already been operating for one year with 28 beams through the first e beam amplifier. The promise of this system lies in ultra-uniform irradiation, making use of induced spatial incoherence, a technique developed at NRL. It is expected that the effective RMS non-uniformity on target can be suppressed down to 0.2% for the 4 ns pulse, equivalent to 10 nm surface perturba-
tions. This would allow new frontiers to be explored in high precision low entropy compression experiments relevant to ICF.

3.3. Light ion beam drivers

Light ion beams accelerated in pulsed power diodes offer advantages for ICF owing to their high efficiency, relatively low cost and efficient beam–target coupling. PBFA II is at present the world’s leading machine. As J.P. Quintenz from SNL pointed out, it has generated lithium beams with a focused intensity of $1.4 \pm 0.4$ TW/cm$^2$ and a beam divergence of 22 mrad at peak ion power. The near term goal at SNL is to reach 5 TW/cm$^2$ and ultimately 50 TW/cm$^2$ for shooting high yield ICF targets. This requires decreasing beam divergence and increasing ion beam power. Another area of active research is transport of ion beams through plasma channels.

The light ion beam programme at ILE has concentrated on developing two stage ion diodes and has demonstrated a significant reduction of beam divergence by this method ($52 \pm 18$ mrad to $27 \pm 6$ mrad for a proton beam accelerated by a voltage of 0.6 MV in the first stage and 1.2 MV in the second). Experiments with carbon beams are in progress, and R&D work on efficient and repetitive pulsed power technology such as saturable core magnetic switches and induction adder accelerators was reported.

At Cornell University in Ithaca, New York, basic research on diode performance is being pursued. Work on beam divergence, ion species composition, diode impedance, power coupling behaviour (e.g. parasitic load), and beam transport and focusing was presented. Interesting work was also reported from the Troitsk Institute for Innovation and Fusion Research (TRINITI), Russian Federation, where the Angara-5-1 complex is used for superfast magnetic implosion of liners and Z pinches for ICF and related purposes. Double liners have been imploded by 4 MA discharges on a 90 ns time-scale with the goal of producing thermal X rays by colliding the liner shells. Experiments have resulted in colour temperatures of 100–120 eV and X ray pulse fronts with rise times of 1–3 ns.

3.4. Heavy ion drivers

Heavy ion accelerators play a rather special role in the development of inertial fusion energy. On the one hand, they are considered to be prime candidates for ICF reactors, as was strongly expressed by M. Sluyter in discussing energy perspectives within the US fusion programme. On the other hand, the heavy ion drivers are the least developed when measured in specific energy deposition actually delivered so far by heavy ion beams on targets. The high potential of heavy ion accelerators lies in their high driver efficiency (20–25% is expected), reliability and durability, as well as their ability to be repetitively pulsed. However, the relatively high costs have been a major concern so far in considering a test facility capable of performing significant
target experiments. In particular, in the USA the philosophy has been first to demon­
strate target ignition and burn in single shot laser experiments and to develop heavy
ion drivers within a longer term programme.

Concerning such a longer term programme in Europe, an initiative for a
European study group aimed at an ICF ignition facility with heavy ion beams has
been launched by a number of European laboratories, with the Gesellschaft für
Schwerionenforschung (GSI), Darmstadt, Germany, acting as a co-ordinator. It is
intended that this study group be concerned with the critical issues of (1) the design
of the driver, consisting of an RF linac with storage rings; (2) the design of the
targets, including the means of their production; and (3) the design of the required
reactor chamber. The initiative combines existing national activities in the field of
heavy ion ICF and has grown out of a series of discussion meetings at GSI and at
the European Organization for Nuclear Research (CERN), Geneva. It was recog­
nized that Europe has unique know-how concerning accelerator technology on which
heavy ion ICF can build.

There were two papers concerning recent work on heavy ion drivers in the USA
and in Germany. R. Bangerter reported on the US heavy ion programme which has
concentrated on the induction linac approach and which is being pursued at the
Lawrence Berkeley Laboratory. In particular, recirculating linacs have been actively
considered. Another line of development is the experimental ELISE project, which
consists of a planned 5 MeV multibeam induction linac with a 2 MeV ion injector
that has already been constructed. The efforts at GSI, reported by I. Hofmann, have
focused on the RF linac/storage ring approach, and new results on the feasibility of
a reactor size driver accelerator were presented.

4. OUTLOOK

Inertial fusion research is preparing for first ignition and burn of a fusion target
in about ten years with megajoule glass lasers currently being designed. Key
problems concerning symmetry and stability of target implosion are expected to be
solved in view of the significant progress made in generating smooth laser beams for
both direct and indirect drive, in successful experimental and numerical modell­
ing of the underlying physics, and in developing the appropriate high resolution diagnost­
cics. Inertial fusion energy production is being considered on a longer time-scale,
with heavy ion accelerators seen as a leading driver candidate. It is hoped that inter­
national co-operation made possible by recent declassification steps will speed up ICF
research.
SUMMARY ON NEW DEVICES, REACTORS AND TECHNOLOGY (INCLUDING ITER)

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1. INTRODUCTION

One invited talk and 22 contributed papers on ITER and 27 contributed papers on new devices, reactors and technology were presented at the conference. This summary first gives an overall classification of the presentations made in these sessions and then introduces the major topics of the sessions. Finally, a brief summary is given on the status of the ITER design and of new devices, reactors and technology as covered at the conference. It should be noted that there are many activities on reactor studies and technology other than those presented at the conference. Information on them may be found in the proceedings of the IAEA Technical Committee Meeting and Workshop held in October 1993 [1] and of two fusion reactor technology symposia held in 1994 [2, 3].

The classification of the presentations made in the sessions is given in Table I. It shows that about half of the 23 presentations on ITER deal with the plasma physics design. This indicates that although ITER is two years into the Engineering Design Activities (EDA), there are still large uncertainties and many unresolved issues in physics aspects of the design.

As new devices, JT-60 Super Upgrade (JT-60SU), the Tokamak Physics Experiment (TPX), Ignitor and, in three presentations, compact tokamaks were introduced. These may be regarded as devices complementary to ITER on the way to realizing a fusion DEMO reactor.

As for reactors, ten presentations were made. Four were related to tokamak power reactors, one to stellarator reactor economics feasibility and two to inertial confinement fusion (ICF) reactors, as well as the strategies to develop heavy ion fusion reactors in the United States of America and the European Union.

In the area of technology, 11 miscellaneous presentations were made. The tritium handling experience in the TFTR DT campaign, the superconducting coil development and local island divertor study for the Large Helical Device (LHD), the R&D of in-vessel components and remote maintenance devices for the Fusion Experimental Reactor (FER), a review of helium cooled blanket concepts for power reactors, and progress on negative ion based neutral beam injectors in Japan and the
### TABLE I. CLASSIFICATION AND NUMBER OF PRESENTATIONS

<table>
<thead>
<tr>
<th>Category</th>
<th>Subcategory</th>
<th>Presentations</th>
</tr>
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<tbody>
<tr>
<td>Overall</td>
<td>Status, Outline Design, operational capability</td>
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<td>Iter (23)</td>
<td>Physics</td>
<td>11</td>
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<td></td>
<td>Engineering</td>
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<td>Tokamaks</td>
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<td>Pulsed reactors, DEMO, SSTR(^a) MHD</td>
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<td>Koyo, HYLIFE-II</td>
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<tr>
<td></td>
<td>Accelerators</td>
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<td>Stellarators</td>
<td>LHD magnets, local island divertors</td>
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<td>Lasers</td>
<td>V alloys</td>
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<tr>
<td>Heavy ion beams</td>
<td>IFMIF(^b), VNS, transmutation reactor</td>
<td>3</td>
</tr>
</tbody>
</table>

\(^a\) Steady State Tokamak Reactor.  
\(^b\) International Fusion Materials Irradiation Facility.

EU were presented. The status of vanadium alloy development was reported. Activities concerning a fusion Materials Test Facility (MTF) and a larger Volumetric Neutron Source (VNS) for in-vessel component irradiation tests were presented. Utilization of a small tokamak for the treatment of radioactive waste was also proposed.

2. **ITER**

The ITER EDA Outline Design was introduced for the first time in a comprehensive manner at the conference. Major changes from the ITER Conceptual Design Activities (CDA) design are as follows: lower plasma elongation \( \kappa = 1.55 \), a single null divertor and constant tension toroidal field coil shape. It was explained that these changes follow from the considerations of optimal use of toroidal field coil volume and provision of reliable vertical control. The major parameters of ITER are shown in Table II.
The ITER EDA Outline Design is aimed at fulfilling the objective of demonstrating "controlled ignition and extended burn of deuterium-tritium plasmas, with steady state as an ultimate goal, and to demonstrate the technologies essential to a reactor in an integrated system, and perform integrated testing of the high-heat-flux and nuclear components". ITER is a full ignition, high power (1.5 GW(th)), long pulse (1000 s) machine with reactor relevant wall loading (1 MW/m²), pulsed power duty factor (≤40%) and significant tritium inventory.

Sensitivity studies have shown that the present design has been successful in maximizing physics and engineering performance while minimizing cost and complexity. Design variants with higher elongation (up to $\kappa = 1.7$) failed to offer appreciable cost/performance improvement. Design variants with fewer toroidal field coils (20 rather than 24) allow better access for NBI current drive and in-vessel component maintenance, but entail either increased toroidal field coil size or loss of plasma performance and vertical control reliability.

The physics basis studies have shown that the scale of the ITER device will bring qualitatively new phenomena to experimental fusion research. The energy per unit area associated with the disruption thermal quench can no longer be absorbed by the heat capacity of the divertor/first wall material but instead will cause vaporization of sacrificial material. Furthermore, power flow in the divertor cannot be directly removed by first wall thermal conduction without undue erosion and melting. In the reference design of the divertor, it is proposed to reduce the target load and electron temperature by radiating about 90% of the power along the divertor channel. The results of the computer simulations indicate that operation of a 'detached' plasma in ITER will require significant amounts of impurity radiation in the scrape-off layer and good confinement of the neutrals in the divertor chamber. The feasibility of the divertor concept must be further demonstrated by the physics R&D.

### TABLE II. MAJOR PARAMETERS OF NEW TOKAMAK DEVICES

<table>
<thead>
<tr>
<th></th>
<th>ITER</th>
<th>JT-60SU</th>
<th>TPX</th>
<th>Spherical tokamak, MTF</th>
<th>Ignitor</th>
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</thead>
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<td>Major radius $R$ (m)</td>
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<td>4.8</td>
<td>2.25</td>
<td>0.57</td>
<td>1.32</td>
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<tr>
<td>Plasma radius $a$ (m)</td>
<td>3.0</td>
<td>1.3–1.5</td>
<td>0.50</td>
<td>0.36</td>
<td>0.47</td>
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<tr>
<td>Aspect ratio $A = R/a$</td>
<td>2.7</td>
<td>3.2–4</td>
<td>4.5</td>
<td>1.6</td>
<td>2.8</td>
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<tr>
<td>Elongation $\kappa$</td>
<td>1.55</td>
<td>1.8</td>
<td>1.8</td>
<td>2.3</td>
<td>1.85</td>
</tr>
<tr>
<td>Plasma current $I_p$ (MA)</td>
<td>24</td>
<td>10</td>
<td>2</td>
<td>6.8</td>
<td>12</td>
</tr>
<tr>
<td>Magnetic field on axis $B_T$ (T)</td>
<td>5.72</td>
<td>6.25</td>
<td>4.0</td>
<td>2.3</td>
<td>13</td>
</tr>
<tr>
<td>Fusion fuel</td>
<td>DT</td>
<td>DD (DT?)</td>
<td>DD</td>
<td>DT</td>
<td>DT</td>
</tr>
</tbody>
</table>
The superconducting magnet system for ITER will consist of 24 toroidal field coils of 468 t each, a central solenoid weighing 902 t, and an external set of six poloidal field coils, the largest of which will have a diameter of 30 m. They will require about 1700 t of Nb$_3$Sn strand and operate at a maximum field at the windings of 13 T. A mechanical configuration where the toroidal field coils are bucked on the central solenoid has been selected. An extensive R&D programme focused on the manufacture and testing of the model coils has been initiated.

A minimum auxiliary power of 50 MW is required to ignite the ITER plasma under the current global transport scaling laws, impurity scenarios and operation conditions. The ITER auxiliary heating system should also be designed with capabilities for current profile control and extension of the pulse length and possibly steady state operation. ICRF heating is considered to be the prime auxiliary heating candidate, because it can fulfil ITER requirements with a minimum of technological development. As the alternative candidates, NBI and EC wave heating and current drive systems are proposed. The 50 MW, 1 MeV NBI system to be developed would consist of four 12.5 MW modules. EC waves can be injected into the tokamak at a power density of 50 MW per port through a simple launch structure. A single frequency of 170 GHz should satisfy the ITER requirements for heating and current drive.

The vacuum vessel is a double wall structure with poloidal strengthening ribs. Steel balls are used to fill the voids. The system is directly water cooled, and with the shielding blanket provides sufficient shielding to keep the magnet heat and radiation damage to acceptable levels.

Two options for the first wall have been developed, one in which the cooling is integrated with the shielding blanket, the other in which the cooling is separate. The first wall heat flux is specified to be as high as possible, consistent with both these design options. The first wall is then capable of removing 80% of the $\alpha$ power with a peaking factor of 2.5. In addition to providing a margin against off-normal events, this specification permits experimental flexibility. A change-out from shielding to breeding blanket is foreseen to meet the needs of the extended performance phase.

3. NEW DEVICES

Major parameters of new tokamak devices, including ITER, are compared in Table II. The table shows the similarity of JT-60SU and TPX in aiming at high aspect ratio and relatively high magnetic field to achieve advanced physics and steady state operation. On the other hand, the spherical tokamak and Ignitor are characterized by low aspect ratio and very high magnetic field, respectively.

A conceptual design of the steady state tokamak JT-60SU has been carried out. It aims to establish the integrated basis of physics and technology for steady state tokamak reactors, i.e. to sustain high performance discharges with high confinement,
full current drive, good stability and high efficiency of heat and particle exhaust for 2000 s, much longer than the characteristic time-scales. The design of the toroidal field coils includes the Nb$_3$Al superconductor. The 25–40 cm thick water tank type vacuum vessel made of Ti–6Al–4V alloy and 1% $^{10}$B doped water were chosen for the DD phase to permit access to the vacuum vessel after about one year of cooldown. Further design study is planned for DT operation.

The mission of TPX is to develop the scientific basis for cost competitive, continuously operating tokamak power plants. It is optimized to achieve improved performance through strong plasma shaping, recycling control and current profile shaping, while operating continuously. The design incorporates poloidal field flexibility for a wide range of operation in normalized $\beta$ and internal inductance. Having superconducting poloidal and toroidal field coils, the TPX device itself is capable of continuous operation, although initially auxiliary equipment will limit the pulse length to 1000 s.

The possibility of steady state operation in a tight aspect ratio ($R/a \leq 1.6$) 'spherical' tokamak as a fusion materials test facility or a future power reactor was discussed. The current drive requirements are deduced for MHD stable equilibria including self-consistently calculated neoclassical bootstrap currents and diamagnetic currents. Numerical models indicate that an NBI system can be tailored to drive the required current for powers consistent with a modest enhancement in energy confinement time, for a range of devices from projected near term experiments to a power plant. Examples of steady state operation described are for a moderate size device ($I_p \approx 0.7$ MA), a materials test facility ($I_p \approx 7$ MA) and a 1 GW(e) power plant ($I_p \approx 27$ MA). Key advantages of the steady state spherical tokamak are the relatively small size and capital cost, particularly when the ability to access second stability is exploited. It is stated that very low aspect ratio ($R/a \approx 1.2$) tokamaks utilizing a demountable, normal conducting centre leg for the toroidal field have the possibility to lead to compact and economically competitive fusion power plants.

Developments in the Ignitor programme were presented. The Ignitor experiment has been designed to reach DT burn conditions by ohmic heating alone at high peak densities, approaching $n_0 \approx 10^{21}$ m$^{-3}$, and relatively low peak temperatures, $T_0 = 11$–15 keV. At $T_0 = 11$ keV, ignition is achieved with values of fusion $\alpha$ particle power $P_\alpha \leq 20$ MW. The ICRF heating system with 18 MW is provided as a reserve heating source in order to reach ignition and to accelerate it. The construction of the key components, such as a complete module (1/24th) of the toroidal field coil, has been undertaken.

4. **Reactors**

A reactor study called PULSAR has been carried out to investigate the commercial attractiveness of pulsed tokamak reactors in comparison with steady state tokamak reactors. Particular attention is given to the thermal fatigue issue for the
fusion power core, especially for the first wall and divertor, to operation of the balance of plant in steady state, and to delivering constant electrical power to the grid by using suitable energy storage such as heat storage in the SiC outer shield. The study resulted in a conclusion that the tokamak pulsed reactor is indeed feasible but that the cost of electricity would be 25% more than for the steady state power plant ARIES-I using similar physics and engineering assumptions.

Assuming conservative physics and conventional technology, an ultra-long-pulse tokamak reactor is considered where a major radius $R = 10$ m is necessary to ensure a pulse length longer than 10 h. The impact of advanced plasma physics on the design of the inductively operated pulsed tokamak reactor has been studied. It is found that an ultra-long-pulse operation of several hours or more is possible in the case of a device with a major radius $R = 7.5$ m, and a flexible operation mode to adapt to a large variation in electrical power demand in the course of a day is available.

The stabilities of high $\beta_p$ plasmas with large bootstrap current are investigated as a function of the pressure profile and the current profile in the Steady State Tokamak Reactor (SSTR). It is found that the way to optimize such a plasma is by controlling $q_{\min}$ to be slightly above one ($\sim 1.1$) with a nearly parabolic pressure profile ($\sim (1 - \rho^2)$).

Major parameters of the four tokamak power reactors are compared in Table III. SSTR and ARIES-I are tokamak power reactors based on advanced physics, with large aspect ratio and high magnetic field, achieving steady state operation utilizing a large fraction of bootstrap current for plasma current drive. PULSAR is an optimized pulsed version of ARIES-I based on similar physics and engineering assumptions.

### Table III. Major Parameters of Tokamak Power Reactors

<table>
<thead>
<tr>
<th>Parameter</th>
<th>SSTR</th>
<th>ARIES-I</th>
<th>PULSAR</th>
<th>Advanced pulsed reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius $R$ (m)</td>
<td>7.0</td>
<td>6.75</td>
<td>9.2</td>
<td>7.5</td>
</tr>
<tr>
<td>Plasma radius $a$ (m)</td>
<td>1.7</td>
<td>1.5</td>
<td>2.3</td>
<td>1.875</td>
</tr>
<tr>
<td>Aspect ratio $A = R/a$</td>
<td>4.1</td>
<td>4.5</td>
<td>4</td>
<td>4</td>
</tr>
<tr>
<td>Plasma current $I_p$ (MA)</td>
<td>12.0</td>
<td>10.2</td>
<td>14.0</td>
<td>10.6</td>
</tr>
<tr>
<td>Magnetic field on axis $B_T$ (T)</td>
<td>9</td>
<td>11.3</td>
<td>6.7</td>
<td>7.32</td>
</tr>
<tr>
<td>Maximum magnetic field $B_{T_{\text{max.}}}$ (T)</td>
<td>16</td>
<td>21</td>
<td>12</td>
<td>13</td>
</tr>
<tr>
<td>Fusion power $P_f$ (MW)</td>
<td>3000</td>
<td>1900</td>
<td>3000</td>
<td>2700</td>
</tr>
<tr>
<td>Current drive power $P_{\text{CD}}$ (MW)</td>
<td>60</td>
<td>100</td>
<td>—</td>
<td>—</td>
</tr>
<tr>
<td>Bootstrap current fraction $I_{CD}/I_p$ (%)</td>
<td>75</td>
<td>68</td>
<td>37</td>
<td>70-80</td>
</tr>
</tbody>
</table>
assumptions, but with the plasma current driven inductively in pulses rather than achieving steady state operation by non-inductive means. The magnetic field on axis is low, in consideration of the cyclic stress on the components. The advanced pulsed reactor is also a pulsed reactor design based on advanced physics and also with a low magnetic field. The physics and engineering assumptions seem to be similar to those of PULSAR except for the inductive current drive only of 20–30% of the plasma current.

Stellarators have significant potential advantages over tokamaks for fusion power plants (no disruptions, no current drive and no stability control system). Four different stellarator configurations (a Compact Torsatron, a new Modular Torsatron, Helias and a new Modular Helias-like Heliac) were analysed as fusion power plants and compared with the second stability tokamak ARIES-IV. The stellarators were described to be economically competitive with ARIES-IV for a range of assumptions on confinement, $\beta$ and $\alpha$ particle losses.

A systems analysis of the laser fusion reactor Koyo, driven by a laser diode pumped solid state laser, and an approach to long lived first wall components for a heavy ion beam fusion power plant, HYLIFE-II, were introduced. It is interesting to note that these reactors claimed a cost of electricity comparable to that of present light water reactors (LWRs). It should be noted that these ICF reactor studies are still at a conceptual stage, while LWRs are actually producing electricity. The advantage was stressed of using a liquid wall to protect the solid first wall from neutron damage and activation. This scheme is attractive and one would like to see it demonstrated.

The physical and environmental characteristics of demonstration reactors (DEMOs) that would extrapolate from ITER were calculated. Direct extrapolation from ITER would lead to DEMOs with major radius $R = 8–9$ m, plasma current $I_p = 20–30$ MA and neutron wall load $\Gamma_n = 1.0–1.5$ MW/m$^2$; the resulting radioactive waste would require deep burial following shutdown. Improvements that would arise from basing the extrapolation also on the results of advanced physics, such as will be explored in TPX, would lead to DEMOs with $R = 6–8$ m, $I_p = 15–20$ MA and $\Gamma_n = 1–2$ MW/m$^2$, with SS-316 structural material. Using a V–4Ti–4Cr structure, in addition to advanced physics, would lead to $\Gamma_n \geq 3$ MW/m$^2$ and to DEMOs the radioactive waste from which probably could satisfy near surface burial criteria in the USA.

5. TECHNOLOGY

The tritium handling experience in the high power DT experimental programme on TFTR that began in December 1993 was described. TFTR was designated as a low hazard nuclear facility, which required that the on-site tritium inventory not exceed $5 \times 10^4$ Ci ($1.8 \times 10^{15}$ Bq). This requirement has imposed special tritium handling and inventory/accountability problems, but will become easier to meet when on-site tritium processing becomes possible in the near future. Several unexpected
occurrences were successfully dealt with and the results provide valuable guidance for future tritium handling in fusion machines, for example on eliminating SF$_6$ as an insulating gas for the power systems.

It is important to focus on the maintainability of in-vessel components. A first wall design based on a double wall fail-safe concept has been developed. A model of this first wall has been constructed and its thermal properties tested. A full scale mock-up of a divertor supporting system was also fabricated, and its locking/lifting mechanism has been demonstrated to be reliable. For the connection and disconnection of cooling pipes, the feasibility of internal access for welding and cutting by a CO$_2$ laser beam was demonstrated by the fabrication of a 10 kW prototype processing head.

On the basis of a review of different helium cooled first wall and blanket designs, the solid breeder, SiC composite material option generates the lowest amount of induced radioactivity and afterheat and has the highest temperature capability. When combined with the direct cycle gas turbine system, it has the potential to be the most economical fusion system and can compete with advanced fission reactors. Development of SiC composite material is the critical issue for the feasibility of such a blanket. Fundamental research has begun in addressing the issues of optimized composite materials, irradiation effects, leaktightness and low activation braze materials.

Extra-large superconducting magnets are being constructed for LHD. Bath cooled composite type and supercritical helium force cooled Nb–Ti conductors have been chosen for helical coils and poloidal coils, respectively. New evaluation methods to determine critical current, stability margin and current distribution for a relatively large current (10 kA or more) superconductor were described.

A local island divertor has been proposed for the edge plasma control of LHD. Although the LHD edge plasma will primarily be controlled by a closed full helical divertor, a local island divertor will be used as an alternative. The advantage of the local island divertor over the helical divertor is the technical ease of the hydrogen pumping because of its toroidally localized recycling.

A large negative ion source for the JT-60U negative ion based neutral beam injector (N-NBI) has been built and has performed an initial test of negative ion production. Focused beams have been extracted even at a low energy of 50 keV at the same perveance as the full energy operation. H$^-$ ions of 0.18 A were successfully accelerated up to 400 keV for 1 s from nine apertures of 14 mm diameter at the design value of the current density for the N-NBI. For R&D of negative ion accelerators for fusion reactors such as ITER, a high energy negative ion test facility, the MeV Test Facility, has been constructed. Testing of a prototype accelerator, which is rated to generate a 1 MeV, 1 A H$^-$ ion beam for a 60 s pulse, has been started.

The European negative ion based neutral beam development programme mainly consists of experiments on high energy accelerators and high current D$^+$ sources conducted at CEA Cadarache, France. A 1 MeV, 0.1 A D$^+$ acceleration experiment
is to start operation at the beginning of 1995. This experiment will test the SINGAP, a highly simplified concept of a negative ion electrostatic accelerator. A new pre-accelerator has also been designed which traps nearly 100% of the extracted stray electrons. In the testing of novel accelerator concepts and the capability to operate in deuterium, the European programme is complementary to the Japanese development programme.

The heavy ion inertial fusion programme in the USA, which emphasizes the development of accelerators for fusion power production, was introduced. Target physics research and some elements of fusion chamber development are supported in the much larger ICF programme, a dual purpose (defence and energy) programme. Large accelerators for high energy physics have high pulse rates, good reliability and long life. It is believed that heavy ion accelerators for fusion can have these same characteristics and also high efficiency. The new feature needed for fusion is high peak beam power (greater than 100 TW). This power must be obtained while retaining the beam quality (low emittance is needed to focus the beams onto a small target). The Induction Linac Systems Experiments (ILSE) programme will provide driver scale beams. The combination of the US National Ignition Facility and ILSE results will, in about 2005, provide the basis for designing and building an accelerator that can be upgraded to drive a demonstration power plant.

In Europe, research on heavy ion beam drivers for inertial fusion has focused on RF accelerators and storage rings, which are commonly used in medium and high energy physics. Studies regarding a full reactor driver accelerator as well as an ignition facility are being carried out. The latter is the subject of an initiative on the part of several European laboratories proposing a study on the feasibility of an accelerator leading to ignition.

Vanadium alloys exhibit important advantages as candidate structural materials for fusion first wall and blanket applications. These advantages include fabricability, favourable safety and environmental features, high temperature and high wall load capability, and long lifetime under irradiation. Major concerns regarding the use of vanadium alloys for fusion applications relate to cost, reactivity with air/oxygen and the limited database.

It is widely agreed that the development of materials for fusion systems requires a high flux, 14 MeV neutron source. The EU, Japan, the Russian Federation and the USA have initiated the conceptual design of such a facility. This activity, under the OECD International Energy Agency (IEA) Fusion Materials Agreement, will develop the design for an accelerator based D–Li system, MTF. The device was originally called the International Fusion Materials Irradiation Facility (IFMIF) but was recently renamed. The conceptual design is to be completed by early 1997.

It was explained that a timely development of fusion nuclear technology components for DEMO, e.g. the blanket, requires the construction and operation of a fusion facility in parallel with ITER. A scenario with only ITER was found to lead to an unacceptably high risk and serious and costly delays in developing the breeding blanket for DEMO. The need for a VNS, a facility dedicated to fusion nuclear tech-
nology testing and development, was stressed. Design concepts for a VNS that best serve the testing needs at modest cost were investigated and it is considered that an attractive design envelope for a tokamak VNS has been identified.

A compact tokamak transmutation reactor for treatment of high level waste was presented. The study showed that a 1 GW(e) fusion transmutation reactor can treat the nuclear waste generated by fission power plants with a total output of ~10 GW(e).

6. SUMMARY

ITER

(1) The ITER EDA Outline Design is aimed at fulfilling the objective of demonstrating "controlled ignition and extended burn of deuterium–tritium plasmas, with steady state as an ultimate goal".
(2) The physics design has been shown by sensitivity studies to be near the optimum under the constraint of various limitations such as cost and reliability.
(3) There remain large uncertainties in the divertor database and simulation, so extensive work is still needed.
(4) Further improvements are needed in the blanket, magnet design and safety approaches.
(5) ITER is the most important programme for the fusion community and we should devote our best efforts to attaining its success.

New devices

(1) JT-60SU, TPX, Ignitor and compact tokamaks could be regarded as devices to complement ITER on the way to realizing a DEMO fusion reactor.
(2) Of the devices JT-60SU, TPX and Ignitor, JT-60SU has the greatest capability and Ignitor the least, but with respect to their construction the most progress has been made for Ignitor and the least for JT-60SU. There seem to be many similarities between JT-60SU and TPX, which aim at advanced physics and steady state operation.
(3) Low aspect ratio tokamaks could be strong candidates for a VNS and could even be extrapolated to be power reactors.

Reactors

(1) PULSAR and the advanced pulsed tokamak reactor designs showed that a tokamak power reactor can be feasible even with a pulsed operation mode, but at some increase in cost.
(2) The stellarator reactors and ICF reactors were claimed to be competitive with tokamak reactors.

(3) In the design of the heavy ion beam reactor HYLIFE-II, a Flibe molten lithium salt layer is effective in protecting the solid metal wall from shock wave and neutron damage and at the same time in reducing the induced activation.

(4) For a tokamak to be an attractive power reactor, it was shown that the achievement of advanced physics and the realization of advanced materials are needed.

Technology

(1) The successful tritium handling experience in the TFTR DT campaign was presented, with some valuable recommendations for the future.

(2) Extra-large superconducting coils are being fabricated and a local island divertor is proposed for LHD.

(3) The R&D of in-vessel components and remote maintenance devices is making steady progress.

(4) Helium cooled blanket concepts for power reactors will be attractive if SiC composite materials can be developed.

(5) Negative ion based neutral beam injectors are progressing in both Japan and France.

(6) The development status and attractive features of vanadium alloys were presented.

(7) The conceptual design of an MTF has been started within the IEA framework and is to be completed by early 1997. A feasibility study for a VNS has also begun under the IEA. For the timely realization of a DEMO fusion plant, these irradiation facilities will be needed in addition to ITER.

(8) Utilization of a small tokamak for treatment of radioactive material was proposed by researchers from China.

REFERENCES


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