MEETING REPORT

SUMMARY OF THE 17TH CONFERENCE ON CONTROLLED FUSION AND PLASMA HEATING, AMSTERDAM, THE NETHERLANDS, JUNE 25–29, 1990

INTRODUCTION

This conference was organized by the FOM-Instituut voor Plasmafysica Rijnhuizen and sponsored by the Plasma Physics Division of the European Physical Society. It was attended by ~500 scientists. The program included 18 invited lectures; from the accepted contributed papers, 24 were selected for oral presentation and 470 for poster presentation. The contributed papers are published as four-page extended abstracts in four volumes in the Europhysics Conference Abstract Series. The invited papers will be published in a special issue of *Plasma Physics and Controlled Fusion*.

The topics presented at the conference were as follows:

1. tokamaks (110 contributions)
2. stellarators (27)
3. alternative magnet confinement schemes (35)
4. magnetic confinement theory and modeling (83)
5. heating by neutral beam injection (NBI) (4)
6. radio-frequency (rf) heating (54)
7. current drive and profile control (40)
8. impurity and edge physics (37)
9. diagnostics (62)
10. basic collisionless plasma physics (34)
11. inertial confinement fusion (10).

This summary covers mainly those topics of greatest interest to *Fusion Technology* readers; theoretical contributions are given minimal coverage.

NEUTRON SOURCES

Ryutov's invited lecture on a new mirror-type neutron source was slightly outside the scope of the conference. He described the need for a neutron source since the typical fluxes of the primary (14-MeV) neutrons at the first wall of a stationary fusion reactor are expected to be in the range of 3 MW/m^2. For a 10-yr first-wall lifetime, the fluence will be of the order of 30 $MW \cdot yr/m^2$. At present, there is practically no systematic information on the behavior of the materials under such conditions. Therefore, it is highly desirable to create a dedicated neutron source for testing purposes. The neutron spectrum produced by fission reactors (even the fast breeders) is much too "soft." Furthermore, the next-step tokamak [e.g., International Thermonuclear Experimental Reactor (ITER)] will provide a fluence of only 3 $MW \cdot yr/m^2$ until the end of its lifetime, around 2020. The lecture stated that it is possible to build a compact, mirror-type neutron source with fluxes from 2 to 4 MW/m^2 with an area of 0.5 to 1 m^2. Both capital and operational costs will most probably not exceed a few percent of the costs of ITER.

TOKAMAKS

The general progress in tokamak research in the past year was reviewed in a number of invited lectures.

Joint European Torus

Dietz reported on the effect of beryllium on the plasma performance in the Joint European Torus (JET). Experiments were carried out initially with a thin layer of beryllium evaporated onto the walls of the machine. Later, the graphite material of the belt limiters was also substituted with beryllium. The major improvements were higher plasma purity levels and higher density operation due to the elimination of oxygen and the strong wall pumping capability of beryllium. Because of the widened operation regime, low-density, high ion temperature discharges on the limiter were possible. Beta limits could be explored and rf-only H modes were obtained.

Despite very accurate alignment of the belt limiter, hot spots were found and local melting of the limiter surface occurred. Consequently, plasma-facing components have to be designed in such a way that alignment is not critical and the heat load is distributed evenly. Further steps are being undertaken to improve the performance.

In his review on fusion-relevant performance in JET, Gibson stated that JET routinely produces plasmas whose main parameters lie within the reactor regime. The density limit for additionally heated discharges in JET with beryllium limiters in clean conditions could be doubled (compared to carbon) up to $nRq_{cyl}/B_T = 33 \times 10^{19}$ m$^{-2} \cdot$T^{-1}. This limit is determined by edge radiation loss processes. Furthermore, a big advantage is the nondisruptive nature of the density limit. The high value of the density limit in JET means that an acceptable limit might be reached in next-step devices provided that a sufficient degree of impurity exclusion can be obtained.

By operating at low toroidal field (TF) (0.9 T), it was demonstrated that the Troyon beta limit could be obtained with an applied power of 10 MW. The introduction of beryllium antenna screens has solved previous ion cyclotron resonance heating (ICRH) impurity production problems. Consequently, effective ICRH is now possible in both limiter and H-mode discharges, reaching temperatures up to $T_i(0) =$ 28 keV and $T_e(0) = 12$ keV. Energetic particles with both isotropic and transverse velocities are contained for at least 2 s while they slow down by apparently classical processes.

A fusion power of 100 kW in charged particles has been produced in D-^{3}He plasmas, and $n_D \cdot T_i(0) \cdot \tau_E$ values up to 9×10^{20} m$^{-3} \cdot$keV$\cdot$s with $Q_{D\text{-}D} > 2 \times 10^{-3}$ have been produced in deuterium-deuterium plasmas. For the same plasma parameters in a deuterium-tritium plasma, this is equivalent to $Q_{D\text{-}T} = 0.7$, $P_\alpha = 2.4$ MW, $P_{fusion} = 12$ MW, and $P_\alpha/P_{loss} >$ 20%. A $Q_{D\text{-}T}$ value above unity would have been achieved in case the H mode should have lasted for another second.

The highest performance conditions in JET are obtained in X-point and H-mode discharges. The duration of these fusion-relevant conditions is limited to ~1 s by a large influx of carbon impurity released from local hot spots on the X-point dump plates. This behavior limits the performance of JET and is a serious problem that must be overcome before a long-burn ignition experiment can be designed.

For operation in 1990, a lower hybrid heating (LHH) system is installed and the carbon tiles on the lower cross-point target plates and the nickel antenna screens are replaced with beryllium. Later, the cross-point target plates will be water cooled and the alignment will be improved. Also, disruption feedback coils will be installed to stabilize $m = 2$, $n = 1$ modes. Early in 1992, installation of an axisymmetric internal pumped divertor is proposed to control the backflow of impurities from the target plates toward the main plasma.

JT-60

In his review of JT-60, which was shut down in October 1989, Ushigusa (Japan Atomic Energy Research Institute) concentrated on the high-power lower hybrid current drive (LHCD) and LHH experiments. A high current drive efficiency of $\eta_{cd} = (2 \text{ to } 3.4) \times 10^{19}$ m$^{-2} \cdot$MA/MW was observed. The confinement time of LHCD plasmas decreased with power in the same manner as the L-mode scaling; however, it was independent of the plasma current. Up to 2.5 V$\cdot$s of the flux was saved by 2 MW of LHCD during 2.5 s. The H mode has been obtained in LHCD-only limiter discharges. Nearly steady-state H mode without edge localized modes (ELMs) was demonstrated for a period of 3.3 s without significant impurity accumulation.

Different frequencies of lower hybrid (LH) waves simultaneously decreased the power threshold above which the H mode could be attained. A strongly peaked temperature profile and the improvement of energy confinement near the plasma center were observed in sawtooth-suppressed neutral beam heated plasmas by LHCD.

In heating experiments, at $I_p = 2.75$ MA, the stored energy increased linearly with heating power of up to 9 MW at $\langle n_e \rangle = 7 \times 10^{19}$ m^{-3}. From these results, Ushigusa concluded that LH waves can be expected to be an attractive tool to heat the plasma and to drive the noninductive current in future high-current tokamaks.

In the JT-60U tokamak ($R = 3.4$ m, $I_p = 6$ MA), which will go into operation in April 1991, the same TF coils will be used as in JT-60, although the volume will be increased by a factor of 3 to 100 m^3. For additional heating, 40 MW of NBI, 15 MW of LH, and 8 MW of ICRH will be installed. From 1994 on, 5 MW of 110-GHz electron cyclotron resonance heating (ECRH) power will be connected; however, Ushigusa could not clearly state which company would be able to deliver the required gyrotrons.

Doublet III-D

Luxon (General Atomics) reviewed the results from Doublet-III ($R = 1.67$ m, $a = 0.67$ m) and the implications for next-generation tokamaks. Beta values as high as 10.7% have been demonstrated in plasmas with reduced toroidal field and values of 5.2% at full field (2.2 T). In general, the maximum attainable beta values were limited by confinement and not stability. The observed beta limits for stability are found to be consistent with accepted theory.

Loss of vertical stability, coupled with subsequent current quench as part of a disruption, has been identified as producing high currents flowing from the plasma boundary into the vessel wall. These currents result, in turn, in higher forces on the vessel than expected from inductive models for current transfer from the plasma to the walls.

The dependence of confinement on physical size has been directly assessed in collaboration with JET with the result that the global energy confinement time, in the interval following the H-mode transition and prior to the first ELM, increases with major radius as $R^{1.5}$.

Careful management of the ELMs resulted in H-mode discharges with a 10-s duration, demonstrating that H-mode confinement is suitable for future long-pulse devices.

Current drive experiments using 1.2 MW of ECRH power launched from the high-field side of the plasma resulted in 70-kA driven current in a 400-kA discharge. In the near future, the 60-GHz ECRH system will be replaced by 2 MW of 120 GHz; in a later stage, a 5-MW installation is planned at 110 GHz.

Experiments with ion Bernstein wave heating did not result in effective central ion heating.

Tore Supra

The Tore Supra tokamak ($R = 2.38$ m, $a = 0.8$ m) is equipped with 18 superconducting TF coils made of niobium-titanium and cooled by pressurized superfluid helium at 1.8 K. The first plasma was obtained in early April 1988, and the design values of the toroidal field ($B_T = 4.5$ T) and plasma current ($I_p = 1.9$ MA) were reached in December 1989. The machine is designed to achieve quasi-steady-state discharges up to 30 s. This is in excess of the thermal time constants. The complete wall system has to be able to exhaust the full power continuously; therefore, all the elements are actively cooled by pressurized water.

As described by Grosman, the control of the plasma edge is mainly based on the use of pump limiters and an ergodic divertor. The first results show that in such a large carbon tokamak the density control is dominated by the wall effects, which do not allow the pump limiter to control the density effectively, although its global exhaust efficiency is strong.

Plasma fueling can be done by "continuous" pellet injection (100 pellets, 30 Hz), resulting in a density increase up to 4×10^{20} m^{-3}.

A total of 24 MW of additional heating power is foreseen using NBI, ICRH, LH, and ECRH. Currently, a 3.7-GHz LHCD installation has injected 2.3 MW into Tore Supra.

Axially Symmetric Divertor Experiment

Mertens [Axially Symmetric Divertor Experiment (ASDEX) Team] gave an overview of the physics of enhanced confinement with peaked and broad density profiles. For ASDEX, five types of discharges exist with clearly improved particle and energy confinement characteristics—four have peaked density profiles and one, the H mode, has a very broad density profile. The strong density peaking cannot, in general, be explained purely by the modifications to the particle deposition profile and sawtooth dynamics, but implies a change in the particle transport coefficients. The main modification found is the reduction of the diffusion coefficient by a factor of 2 or more. In addition, the particle inward pinch is anomalously large under most conditions. These findings cannot yet be explained by any theoretical model. In his conclusions, Mertens speculated that the H mode and, to a lesser extent, the pellet regime of confinement are some of the most robust regimes that might allow ignition.

In another paper, Loch showed that the experimentally determined penetration depths of pellets in ASDEX are in good agreement with the scaling law based on a simple model.

Tokamak Experiment for Technology Oriented Research

Messiaen gave an overview of the results of additional heating of the Tokamak Experiment for Technology Oriented Research (TEXTOR) tokamak (R = 1.75 m, a = 0.46 m). Different heating schemes were applied: (a) NBI, with two deuterium beams at 55 keV of 1.7 MW each, both in the codirection (NBI-co) and in the counter direction (NBI counter); (b) ICRH, 2.2 MW at 32.5 MHz; and (c) combinations of these heating methods. Stationary heated plasma conditions could be obtained on TEXTOR with I_p = 340 kA, B_T = 2.25 T. Boronization of the wall has decreased the radiation losses substantially. The main results are a significant enhancement of the inductively driven toroidal current (up to 60% of the total current) and of the neutron yield when ICRH is added to NBI-co.

Coupling up to 6 MW of additional power to the TEXTOR plasma has led to a β_p of 1.5%. Complete sawtooth stabilization has been obtained, including stabilization of the "monster" sawteeth. Large enhancements with respect to the Kaye-Goldston scaling were found with NBI-co. Peak electron and ion temperatures around 3 keV were obtained.

TEXT

In TEXT (Rowan et al.), the confinement time for particles and impurities is significantly longer in a helium plasma than in either hydrogen or deuterium plasmas. This is largely due to a difference of a factor of 20 in the convective inward velocity.

T-10

At T-10 (R = 1.5 m, a = 0.28 m), the density limit could be enhanced by additional gas puffing during discharges with 1.6 MW of ECRH power at 81 GHz (Alikaev et al.).

Rijnhuizen Tokamak Project

The first experiments with the Rijnhuizen Tokamak Project tokamak (R = 0.72 m, a = 0.18 m), which were devoted to transport studies, were presented by Polman et al. Breakdown plasmas with a current up to 2 kA could be produced and sustained by ECRH power alone.

Scaling Laws

Although there is much effort toward establishing scaling laws, Engelmann's [Next European Torus (NET) Team] conclusion that there does not yet exist a theoretical basis for extrapolation to larger devices was not disputed. A reliable scaling law for the energy confinement time based on dimensionless parameters is especially lacking. The scaling with a/ρ_L is very important, since for a next-step device a/ρ_L is to be increased by a factor of 2 with respect to present large tokamaks. This dependence is the most critical part for the extrapolation. However, several scalings are still possible: $\tau_{E,B} \sim (a/\rho_L)^2$ (the Bohm scaling) or $\tau_{E,GRB} \sim (a/\rho_L)^3$ (the gyro-reduced-Bohm or plateau scaling).

Determination of Transport Coefficients

Lopes Cardozo (FOM-Rijnhuizen) reviewed the studies of transport using perturbation analysis. He explicitly discussed the approximations made in the standard analysis and showed that, in principle, the off-diagonal elements of the transport matrix can be studied. This was illustrated by an analysis of cross coupling between electron thermal and particle transport using simultaneous measurements of heat and density pulses in JET.

NET/ITER

Wégrowe et al. have modeled the plasma current ramp-up by LH waves for ITER (R = 6 m, a = 2.1 m). The conclusion is that 20 to 30 V·s can be saved by launching 20 to 30 MW of LH waves.

The m = 2 tearing mode in ITER can be stabilized by 20 MW of ECRH at a frequency of 120 Hz according to Kuznetsova et al.

Fidone and Giruzzi showed that a feasible heating scenario for NET (R = 6.3 m, a = 2 m) is the use of downshifted frequency using the ordinary mode from the low-field side even for gyrotron frequencies as low as 100 GHz.

Compact Ignition Tokamak

For Compact Ignition Tokamak (R = 2.1 m, a = 0.65 m), a ramp-up scenario is foreseen in which 35 MW of ECRH power at a frequency of 308 GHz is used to get the plasma into the burn phase. During ramp-up, $n_e(0)$ will be increased to 9×10^{20}, I_p to 12 MA, and simultaneously B_T to 11 T (Smith et al.).

Wendelstein VII Advanced Stellarator

Ringler reviewed the results of the confinement studies on the Wendelstein Advanced Stellarator (W VII-AS) ($R = 2$ m, $a = 0.2$ m), which, with its nonplanar modular field coils, has operated routinely since January 1990 with full-field parameters. The W VII-AS has a low-shear magnetic configuration to avoid low-order rational values of the rotational transform throughout the plasma confinement region.

Most of the investigations were devoted to ECRH plasmas at 1.25 T (second-harmonic X-mode launching) and 2.5 T (fundamental O mode). Four 70-GHz gyrotrons with a pulse length of 3 s have been used. The maximum plasma parameters that have been achieved were $T_e(0) = 2.5$ keV, $T_i(0) = 0.4$ keV, $n_e(0) = 5 \times 10^{19}$ m^{-3}, $\tau_E = 10$ to 20 ms, and $W = 8$ kJ. These values are as good as the results with tokamaks, given the size of the stellarator.

Density profiles became hollow in the plasma center with increasing ECRH power (200 to 800 kW). The combination of both NBI and ECRH has already led to a stationary state. Future experiments will also use a 140-GHz pulsed gyrotron for density and impurity control. At this moment, W VII-AS is being carbonized and will be boronized later in 1990 to reduce radiation problems.

Electron cyclotron current drive experiments on a stellarator are very interesting, since the small electron-cyclotron-driven currents are not masked by large inductively driven currents in tokamaks, as shown by Erckmann et al.

Other Stellarators

The review lecture by Lyon (Oak Ridge National Laboratory) on the latest stellarator results in Japan, USSR, and the United States made it clear that a wide range of stellarators is under operation, development, or construction. Most of these have a high shear (with a central magnetic well) in contrast to the low-shear stellarator W VII-AS and heliacs (with a global magnetic well encompassing the entire plasma).

One of the general results with stellarators is that an improved plasma behavior is achieved with a small inward shift of the vacuum magnetic axis. Confinement scaling is similar to that in tokamaks, but the positive density and field dependence offsets the degradation with increasing power. Particle and impurity control is not a major issue at present, but helical divertors will be required for steady-state operation. From both the Compact Helical System (CHS) ($R = 1.0$ m, $a = 0.2$ m) and the Advanced Toroidal Facility (ATF) ($R = 2.1$ m, $a = 0.27$ m), a beta up to 1.5% is reported with an inward shift of 3 cm. At ATF the second stability regime is attained.

A new stellarator is being developed: the Large Helical Device ($R = 4.0$ m, $a = 0.55$ m) with a toroidal magnetic field of 4 T and no pulse length limitation. This stellarator will become operational in 1997.

As an overall conclusion, it was stated that stellarators with significant shear are maturing as a confinement concept. The research is also very relevant for tokamak research. Since higher additional heating powers are required, a 1-MW ICRH installation on ATF is foreseen. Furthermore, a 106-GHz, 0.5-MW ECRH installation is now being planned for both Heliotron-E and ATF, and an additional neutral beamline is planned for ATF and CHS.

ALTERNATIVE MAGNETIC CONFINEMENT SCHEMES

Reversed-Field Pinches

The progress in the understanding of transport and fluctuations in reversed-field pinches (RFPs) was clearly reviewed in an invited paper by Prager (University of Wisconsin). The observations of magnetic fluctuations and the associated transport of current (dynamo effect) are reasonably well explained by theoretical models. The driving force behind the large electrostatic fluctuations in the edge and the cause for the anomalous ion heating are unknown.

The RFP as a reactor concept confronts three major issues: the electrical boundary condition, confinement, and current drive. The coming generation of large machines, MST (0.6 MA, in operation since March 1990), RFX (2 MA, operational in July 1991), and ZTH (4 MA, operational in 1993), will test the reactor feasibility. Preliminary results from MST on confinement scaling are not very encouraging.

Z-Pinch

Shlachter [Los Alamos National Laboratory (LANL)] showed that interesting fusion research is still possible on a low budget by reviewing the high-density Z-pinch (HDZP) work at LANL. In these experiments, a solid deuterium fiber is ohmically heated by a high current on a typical time scale of 100 ns. The final goal is to reach ignition temperatures at solid-state density. In HDZP-I, fibers with a 20-μm cross section were heated by a 300-kA current to a temperature of 350 eV. The neutron yield was on the order of 10^7/pulse. In HDZP-II, which recently came into operation, the preliminary results with a fiber with a 25-μm cross section and a 650-kA current (final value will be 1.2 MA) yield a temperature of 3 keV and a neutron yield of 4×10^9/pulse.

IMPURITY AND EDGE PHYSICS

Magnetic confinement studies have shown that plasma/wall interaction and the properties of the peripheral plasma are important for the quality of the plasma in the confinement region. Control of both the release and transport of impurities, and of deuterium fueling and helium exhaust, is essential for a tokamak reactor.

Two main developments concerning choice and treatment of material for plasma-facing surfaces have led to confined plasmas of higher quality, i.e., plasmas with reduced impurity level and a better fueling control. These developments are the use of beryllium in JET and the boronization procedure developed on TEXTOR and now applied in other tokamaks.

The introduction of beryllium in JET—initially only by evaporation, thus forming a thin coating on the plasma-facing surfaces, and later supplemented by introducing beryllium-tiled belt limiters—has had beneficial effects on the plasma. The oxygen contamination was strongly reduced to a negligible level because of gettering by the beryllium layers to form beryllium oxide. The carbon content also decreased, especially in the beryllium limiter phase. The radiation losses were reduced and Z_{eff} reaches values <1.5. To raise the density, a higher gas influx (because of deuterium pumping by beryllium) was needed that could be better controlled (i.e., no uncontrolled release of deuterium from the walls or limiter occurred during the shot), so that the density limit was increased by a factor of 1.6 to 2 compared to the carbon phase.

In general, this limit is not followed by a hard disruption but rather leads to development of a multifaceted asymmetric radiation from the edge (MARFE) accompanied by a decrease in density and return to a quiescent discharge. The electron temperature in the scrape-off layer (SOL), to be influenced by the gas puff rate, was ~30 eV and reduced to 10 eV when MARFE occurred. It seems that the ion temperature in the SOL is much higher than the electron temperature, which could lead to a higher sputtering rate. A major disadvantage of beryllium-tiled limiters is the rather low melting point (1300°C), so that sweeping of the hot spot seems necessary at high power loading to avoid carbon/beryllium blooming. Nevertheless, in the Tokamak Fusion Test Reactor (TFTR) and in JET, carbon blooms have been observed when excessive localized heating occurs on the carbon limiters resulting in a huge carbon release and thereby an increase in radiation losses. Note that the use of the highest heating powers available in TFTR is limited.

A second reported method of wall treatment is boronization. Boronization of plasma-facing surfaces also leads to better plasma performance. This technique (already in use on TEXTOR and ASDEX) has been carried out on TFTR and on TCA. A film of boron and carbon seems highly resistant to chemical erosion and can getter the oxygen. Boronization is achieved with a glow discharge in helium with an admixture of CH_4 and B_2H_6. The main benefit is the ability to getter oxygen, but also metal contaminations are drastically reduced. The radiation losses were reduced in all cases studied and a higher density limit in TFTR and ASDEX, and a longer pulse duration in TCA, could be achieved. In ASDEX, when improved confinement is achieved, metallic impurities may accumulate toward the center while light impurities do not, under both carbonized and noncarbonized conditions. Under boronized conditions, light impurities accumulate as well, and markedly peaked Z_{eff} profiles were shown for counter-neutral-beam heated discharges.

Critical issues for the divertor design in NET/ITER-like devices are the peak heat loads and erosion (Borrass). Values of up to 30 eV for the shear edge temperature would be marginally acceptable for graphite or tungsten first walls.

DIAGNOSTICS

Reflectometry has now become a standard technique for the measurement of density profiles. Prentice et al. (JET Team) showed the first results of a narrow-band reflectometer on JET with a 1-mm resolution.

Active beam scattering will be used to measure the ion temperature at TEXTOR (Barbian et al.). By means of a time-of-flight analyzer, the energy of the particles is measured. Both 3- and 14.7-MeV protons are measured in Tore Supra to obtain information on the fusion profile (Martin et al.).

At ASDEX, a two-point correlation analysis with probes, normally used to study low-frequency turbulence, is applied to investigate also the wave number spectra from the high-frequency waves launched into the plasma by means of the LH grill antenna (Krämer et al.).

INERTIAL CONFINEMENT FUSION

Discussion of inertial confinement fusion was only a minor topic, because this alternative approach to fusion is not extensively studied in Europe. Nakai (Osaka University) discussed the recent results of the GEKKO XII experiment. The most important improvements are the reduction of the nonuniformity of the radiation (by introducing random-phase plates in each of the 12 laser beams) and a better target uniformity. The latter is achieved by improved density matching in the so-called water/oil/water emulsion method. In this way, the deviation from a perfectly spherical shape and nonuniformity of the target thickness were controlled within 1%. With a 1-ns Gaussian pulse of 10 kJ at $\lambda = 0.53$ mm, a density of 600 times solid density has been achieved. It is now expected that to reach ignition, 100-kJ pulses are necessary.

The plans for Nova upgrade were also presented. The pulse energy will be increased from 45 kJ in 2 ns to 2 MJ in 3 ns. This will cost about $400 million and should result in ignition around the year 2000.

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