MEETING REPORT

SUMMARY OF THE 16TH INTERNATIONAL ATOMIC ENERGY AGENCY CONFERENCE ON CONTROLLED FUSION RESEARCH, MONTREAL, CANADA, OCTOBER 7–11, 1996

This conference on controlled fusion energy research is hosted every 2 yr by the International Atomic Energy Agency. The primary emphasis is to overview the physics and to some extent the relevant technology accomplishments of the preceding 2 yr. The 16th conference was held in Montreal, Canada, October 7–11, 1996. A total of 305 papers were presented, all of which will be published in a special issue of *Plasma Physics and Controlled Fusion Research* after a review process (July to December 1997). Papers presented were in the areas of tokamaks, inertial confinement fusion (ICF), alternate magnetic confinement concepts, and the International Thermonuclear Experimental Reactor (ITER). The number of papers presented and attendance at this conference are restricted to a fixed number of participants from each country, which is determined by the extent of fusion research conducted by that country. Thus, a large number of papers were presented by the United States (94), Japan (75), Germany (27), Russia (15), the European Community (14), France (11), the United Kingdom (9), and Canada (4). In addition, 32 papers were presented by the ITER project alone, which reflects the seriousness of this project. Significant experimental time has been devoted to large tokamaks to address the remaining issues for ITER physics and technology.

This summary presents the topics as follows: tokamaks, confinement, advanced confinement regimes, power threshold, beta limits, divertors, helium removal, fast shutdown, halo currents, density limit, next-step devices, the spherical tokamak, high beta, current drive, stellarators and torsatrons/heliotrons, reversed-field pinch (RFP), field-reversed configuration (FRC), spheromak and tandem mirror, ICF, beam interaction with plasma, filamentation and target ripple, indirect drive, improved targets, beam smoothing, light-ion beams, and heavy-ion beams.

TOKAMAKS

A tokamak is a toroidal plasma confinement system in which a toroidally driven plasma current generates a poloidal magnetic field that confines the plasma. In addition, toroidal and vertical magnetic fields are generated by external coils to provide a stable equilibrium. The toroidal field (TF) is much higher than the poloidal field (PF), which results in low values for the Troyon beta. Tokamak physics has developed to the point where it is now possible to build a reactor-sized device to demonstrate reactor-relevant plasma operation and to advance fusion engineering to the point that would allow for the construction of a net power-producing reactor. A near-term goal of the fusion program is to build ITER to carry the tokamak concept forward to demonstrate ignition physics and to develop fusion reactor engineering. The goal for ITER is to demonstrate controlled ignition and extended burn in inductive pulses with a duration of $\sim$1000 s at an average neutron wall loading of $\sim$1 MW/m^2. In addition, demonstration of the necessary technology for a fusion reactor is planned for ITER. The status of tokamak research will be discussed in the context of the ITER project.

Confinement

For a reactor to be ignited, the alpha-particle heating power must exceed the energy loss rate. Early empirical tokamak scalings for an ohmically heated discharge such as the neo-Alcator scaling showed τ_E (the energy confinement time) to have a liner density dependence (up to a saturation level). For discharges with auxiliary heating, the confinement was found to degrade with heating power, and τ_E for such discharges was summarized in the ITER89P L-mode scaling, which showed a weak density dependence. A revised scaling based on data from 14 tokamaks was presented at this meeting (ITER96P) and shows a stronger density dependence but also stronger power degradation. Both these scalings show Bohm-type confinement and result in τ_E of $\sim$2.2 s for the ITER parameters. For ITER to ignite, τ_E should be $>2.5 \times \tau_E^{\mathrm{ITER96P}}$. Thus, the edge-localized mode (ELMy) H mode is considered to obtain sustained ignition. The ITER93, ELM-free confinement includes data from 7 tokamaks and is now being revised to include data from 11 tokamaks [ASDEX, ASDEX-U, Compass-D, Alcator C-Mod, Doublet-IIID (DIII-D), Joint European Torus (JET), JFT-2M, JT-60U, Princeton Beta Experiment-Modification (PBX-M), Poloidal Divertor Experiment (PDX) and Tokamak Configuration Variable (TCV)].

At the reference conditions on plasma confinement, ITER has an $\sim$10% margin in τ_E. At 1.5 GW of fusion power, ITER must operate above the Greenwald density limit unless the energy confinement will be $\sim$20% better than predicted by ITER reference τ_E scaling; $n_{Greenwald} = I_p/(\pi a^2)\,[(10^{20}\ \mathrm{m^{-3}} \cdot \mathrm{MA} \cdot \mathrm{m})]$ describes the observed density limit, but it is not known if this applies to a large-sized device such as ITER.

The scaling for the ELM-free H mode suggests a gyro-Bohm scaling [$\chi \propto \chi_B(\rho/R)$, where $\chi_B \propto T/eB$ and ρ is the gyroradius]. Present tokamaks are able to obtain similar values of the normalized plasma pressure β ($\beta \propto Tn_e/B_T^2$) and the normalized collisionality ν^* ($\nu^* \sim qn/T^2$), but the normalized gyroradius ρ^* [$\rho^* = \rho_g/a \sim (A_iT)^{0.5}/(aB)$] will be smaller in ITER. Detailed experiments are being carried out on JET in which each parameter is changed while the other two are kept fixed. Initial results indicate a favorable dependence on β that, if confirmed by other machines, would increase ITER confinement at ignition by 10 to 15%.

Advanced Confinement Regimes

Several new operating modes have been discovered that may allow ITER to operate at higher values of τ_E. A new confinement mode called negative central shear (NCS) or enhanced reverse shear (ERS) has been obtained [JET, DIII-D, JT-60U, Tokamak Fusion Test Reactor (TFTR)]. A reversed-shear configuration has $-ve$ magnetic shear in the inner region (inside the inner transport barrier) and $+ve$ magnetic shear in the outer region. It is generally produced by injecting a neutral beam (NB) during the current rampup phase. High T_e (due to NB heating) retards the current penetration and helps the formation of a hollow current profile necessary for the reversed-shear configuration. The formation of an internal transport barrier in the ERS regime dramatically reduces the ion heat and particle flux from the core.

NCS discharges are characterized by improved performance. On DIII-D, β_N reached 4.0 ($\beta_T \sim 6.7\%$) while the confinement remained very high, $H \sim 4.5$ and $\tau_E \sim 0.4$ s ($\beta = 2\mu_0 p/B^2$, normalized beta $\beta_N = \beta\, aB_T/I_P$ and $H = \tau_E/\tau_{\mathrm{ITER89P}}$). The central beta reached an impressive 44%.

Because the hollow current profiles generated using NB during current rampup would decay into normal (positive) shear discharges on a resistive timescale, for this mode to be reactor relevant, the negative shear must be maintained by noninductive current drive in combination with bootstrap current drive. DIII-D demonstrated that fast-wave current drive can help to control the current profile in such discharges. On JT-60U, lower-hybrid current drive was used to sustain the reversed-shear mode.

On ASDEX-U, the completely detached H mode (CDH) was produced by neon injection in the divertor, which resulted in a large radiated power fraction and a detached divertor plasma while maintaining H mode confinement. The heat load to the divertor was significantly reduced, but the Z_{eff} increased.

On TFTR, in general, deuterium-tritium (D-T) plasmas show improved confinement compared to similar deuterium plasmas. The supershot regime, which is characterized by peaked density and pressure profiles and confinement enhancements up to approximately three times ITER89P scaling (obtained by maintaining low edge recycling and neutral beam deposition in the plasma center), resulted in fusion performance to 10.7 MW of peak power and 6.5 MJ of fusion energy per pulse (at $B_T = 5.5$ T, $I_p = 2.7$ MA, and $P_{\mathrm{NB}} = 40$ MW).

On the Tokamak Experiment for Technology Oriented Research (TEXTOR), the radiative improved mode (RI mode) was obtained in quasi-stationary discharges (in a pumped limiter configuration) with strong additional heating in the presence of NB coinjection and edge radiation cooling using neon edge impurity seeding. Densities close to the Greenwald limit were attained with confinement similar to the ELM-free H mode with low edge q. There were no ELMs, and heat was dissipated by edge radiation cooling. Impurity seeding did not seem to have a negative effect on fusion reactivity, which was maintained for $100\tau_E$.

The H mode confinement on Alcator C-Mod was found to exceed the recent empirical H mode scaling by a factor of 1.5. Whether this is because of better confinement or a result of inaccuracies in the empirical scaling due to machine size, variation has an important implication to ITER and needs to be examined further.

On JT-60U, with improved core confinement, the highest fusion triple product was recorded in the high β_p H mode at $T_i(0) = 45$ keV [$nD(0)\tau_E\, T_i(0) = 0.51 \times 10^{21}$ keV·s/m^3, with $W_{diamagnetic} = 9.32$ MJ for 0.5 s ($\sim 1.5\tau_E$)] with ELM activities. A 0.5-MeV, 10-MW $-ve$ ion–based NB has begun operation that will extend NB injection (NBI) into reactor-relevant plasmas.

Power Threshold

A heating power exceeding a threshold power P_{thr} is necessary for the L-H transition after other conditions are satisfied. ITER's scaling for this predicts 100 ± 2 MW for ITER at $n_e = 0.5 \times 10^{19}$ m^{-3}, while the JT-60U scaling predicts 53 MW.

However, data from Compass-D and TCV predict higher thresholds; it is not known if this is because these two devices are more sensitive to neutrals, which increases P_{thr} as documented by JT-60U (as these are small devices with open divertors). Alcator C-Mod, which is physically similar in size to Compass-D, has a low P_{thr}. Experiments on DIII-D and JT-60U show that P_{thr} has a minimum for densities in the middle of the operating space and increases for densities higher or lower than this. This has been studied in detail for JT-60U, and it has been shown that there is a low-density boundary below which the L-H transition does not occur. JET and DIII-D scalings indicate that P_{thr} scales as $S^{0.5}$, where S is the surface area at the separatrix. These issues are under further study including edge neutral measurements on Compass-D and TCV.

Beta Limits

The target operating point for ITER is at β_N of 2.5, which is well below the ideal magnetohydrodynamic (MHD) β_N limit of 3.5 for ITER. The ideal MHD beta limit is given as $\beta(\%) \leq 4l_iI/aB$ MA/m·T^{-1} (l_i is the internal inductance). While present tokamaks have exceeded these parameters, they have done so at collisionality higher than that at which ITER will operate. At ITER-like collisionality (and long-duration discharges, with fully relaxed current profile), the attainable β_N appears to be ~ 2.5, and this is accompanied by island formation and confinement degradation. At low collisionality, a soft beta limit exists, which results in degradation of τ_E as observed in several tokamaks (ASDEX-U, DIII-D, and JT-60U at β_N of 1.8 to 2.5 and ITER-type collisionality). While this requires further study, it is encouraging to note that advanced confinement regimes such as the NCS configuration on DIII-D have demonstrated β_N of 6 and an H factor of 4 simultaneously.

Divertors

The purpose of divertors is to reduce plasma/surface interactions by exhausting the used plasma via a magnetic divertor to a chamber where it can be neutralized and pumped out. New tokamaks operate with a divertor. Under ITER operating conditions, the heat load to the divertor plates will be large enough (30 MW/m^2) to erode the target plates. Thus, a radiative divertor, one in which a large amount of exhaust heat is radiated some distance away from the target plates (detached operation), has been under study for many tokamaks because this would reduce the heat load to the target plates to ~5 to 10 MW/m^2. During detachment, the ion flux to the divertor target plates decreases, while the density is increased by gas fueling.

Many tokamaks have obtained fully detached operation of the divertor. Detachment usually occurs at some critical density below which there is no detachment. On Alcator C-Mod, detached H mode operation was obtained by N_2 puffing. The CDH mode on ASDEX-U was obtained by neon seeding. Some form of impurity seeding to obtain detachment will be used for ITER. A large portion of JET experimental time has been devoted to optimizing divertors under ITER-relevant operating conditions. On JET, during 1995, a relatively open (to neutrals) MkI divertor and in-vessel cryopump were used. For high power handling, the magnetic configuration was swept (4 Hz) horizontally with the help of the in-vessel divertor coils (to spread the energy over a large area of the target plates). This allowed energies in excess of 180 MJ [carbon-fiber composite (CFC) tiles] and 120 MJ (beryllium tiles) to be conducted to the tiles without the occurrence of significant sublimation or melting. The value E_p reached 13.5 MJ. The steady-state H mode with a radiative divertor using N_2 as the seed gas was studied. In 1996, Mk2, a relatively closed divertor, with continuous water cooling was installed. More flux-wetted area should allow power-handling factors ~3 to 5 times that of MkI. Operation is underway in which the flux strike points are chosen to be on either the horizontal or the vertical plates to determine the effect of plate orientation as well as effects on impurity retention and neutral recycling.

The goal for the component lifetime in ITER is 3000 full-power discharges, including up to ~10% full-power disruptions and 10% transient power excursions (~20 MW/m^2, up to 10 s of "attached" operation). Thus, the choice of target plate becomes important. Present tokamaks use beryllium or graphite plates. Graphite has a large hydrogen retention capability, which might lead to an unacceptable tritium inventory, while beryllium suffers from a high physical sputtering yield as well as from a low melting point and a high vapor pressure. Therefore, the use of tungsten is planned for many parts of the ITER divertor plates because tungsten exhibits much more resistance to erosion and hydrogen retention. But, the high self-sputtering of tungsten means that the temperature must be <75 eV. Also, tungsten concentration in the main plasma must be $<10^{-5}$.

On ASDEX-U, almost 90% of the plasma-facing surface in the strike zone was covered by tungsten. In most cases, under normal discharge conditions, the tungsten concentration in the main plasma was close to, or even lower than, 10^{-5}. Power load on the tungsten plates reached 6 MW/m^2 under stationary conditions and peak loads of up to 15 MW/m^2 during ELMs in low-density H mode discharges. Tokamak de Varennes (TdeV) researchers are in the process of installing tungsten target plates on the lower divertor, while using CFC on the top divertor so that a comparison could be obtained under similar operating conditions.

Tore Supra, a limiter tokamak, has installed a new type of divertor called an ergodic divertor to control plasma/wall interaction.

Fully detached plasmas (with high neutral pressure) show degraded H mode confinement, while those with impurity seeding or semidetached operation show increased Z_{eff}. This may be due to unoptimized divertor designs that lead to leaks of neutrals due to looser baffles; thus, with many tokamaks (JET in particular), there is a concerted effort to provide tighter baffling to improve and optimize divertor operation.

Helium Removal

For ITER, the helium ash produced by the fusion reactions must be removed at a rate of 2 $Pa \cdot m^3/s$ so that the core helium concentration can be maintained below 10%. Calculations show that this translates to a global helium confinement of <7 to 15 energy confinement times ($\tau^*_{He}/\tau_E <$ 7 to 15), depending on the impurity content of the plasma. Also, present experiments show the helium enrichment in the divertor to be >0.2 (relative to the separatrix), which is adopted as the requirement for the helium concentration in the divertor of ITER.

To simulate ITER-like conditions in DIII-D, a helium NB was used to fuel the core. A divertor cryopump was used to exhaust the helium; $\tau^*_{He}/\tau_E \sim 8.5$ was obtained. The divertor helium enrichment was between 0.2 and 0.6. The exhaust rate seemed to strongly depend on the divertor pumping rate. In addition, helium transport studies on DIII-D show $D_{He}/\chi_{eff} \sim 1$ in all confinement regimes to date (with little dependence on the normalized gyroradius), while the steady-state helium density profile has the same shape as the electron density shape, which suggests that preferential helium accumulation should not be a problem with ITER. Because these results were obtained for nondetached plasmas, whether such a high level of helium retention can be obtained for a detached plasma needs to be determined. In a detached plasma, the mean-free-path against ionization of the helium neutral recycling from the divertor is expected to be long, leading to a possible reduction of the retention time in the divertor. However, it is believed that through a combination of gas puffing and divertor pumping, scrape-off-layer (SOL) flow could be generated, which can be used to increase helium exhaust, while maintaining the desired core density. Helium exhaust in ASDEX-U was found to improve with increasing divertor neutral flux density, and in the H mode and the radiative CDH (fully detached), the enrichment in the divertor was >0.2. The nominal ITER gas fueling rate is 50 $Pa \cdot m^3/s$ (for replacing D-T burnup), while up to 200 $Pa \cdot m^3/s$ is allowed for to generate SOL flow for flushing out impurities (and helium), and as much as 500 $Pa \cdot m^3/s$ is available for 50 s.

Fast Shutdown

Single-null plasma configurations such as will be used in ITER are susceptible to vertical displacement events (VDEs), which can lead to a disruption. To reduce divertor erosion, one should have the means to rapidly remove the thermal and magnetic energy before the plasma moves a significant distance (<3 s for ITER).

In Alcator C-Mod, this requires a dissipation of 1 MJ in 500 μs. In addition, T_e must be reduced to <10 eV so that the plasma inductance over resistance time can be reduced to provide the necessary 0.5-ms quench times. Injection of plastic pellets impregnated with 2.5 mg of silver powder resulted in the quench time being reduced from 5 to 1 ms. Such quenching was observed to be accompanied by a short burst of hard X rays at the time of pellet ablation (also seen with other tokamaks), which indicates the probable generation of runaway electrons. Calculations for ITER show that small amounts of impurities can consume the thermal energy in <1 s, but because of the large amount of magnetic energy that is available in ITER, runaway electrons would be an issue.

On JT-60U, fast current shutdown by killer pellet injection was demonstrated without generating harmful runaway electrons. This was accomplished by a set of external helical coils that were pulsed $\sim$500 ms before pellet injection. The magnetic "burst" fluctuation eliminated the superthermal electrons.

Another scheme proposed by ITER is the massive injection of D_2 or helium of more than 50 g to dissipate the stored magnetic energy in <1 s without leading simultaneously to a large runaway current.

Halo Currents

During the last 2 yr, toroidal asymmetry of the halo current was observed on many tokamaks during VDEs (Alcator C-Mod, ASDEX-U, Compass-D, DIII-D, JET, and JT-60U). Halo currents are currents that flow from the plasma into the vessel wall and back into the plasma. The toroidal asymmetry can create large mechanical forces on the ITER shield and vacuum vessel. In addition VDEs can also create eddy currents, which results from the inductive response of the wall to changes in plasma current or position.

On Compass-D, poloidal and toroidal halo currents during VDEs were studied. The creation of locked modes of various phases prior to the VDE had no effect on the halo currents, which suggests that the initial conditions of the disruption are less important than the dynamics during the quench phase.

In Alcator C-Mod, halo currents are found to be toroidally asymmetric, with peaking factors (peak/average) typically 2 to 3. The asymmetric pattern usually rotates toroidally at a few kilohertz, thus ruling out first-wall nonuniformity as the cause of the asymmetry. In Alcator C-Mod, the maximum halo current during a disruption scales as $\sim I_p^2/B_\varphi$ or I_p/q_{95}.

Density Limit

Experiments to date have not been able to exceed the Greenwald density limit while maintaining nondegraded H-mode confinement. The reference ITER operational scenario requires $n_e \sim 1.1$ to $1.5 \times n_{Greenwald}$. ASDEX-U demonstrated $n_e \sim 1.5 \times n_{Greenwald}$ under degraded H mode conditions; similar results were obtained on DIII-D. While the Greenwald scaling is not fully understood, it is thought that this may be related to an edge radiation limit. Deep fueling has been suggested as a means to overcome this limit.

In addition to the gas injection system, ITER will use two centrifugal pellet injectors designed to inject up to 100 $\text{Pa}\cdot\text{m}^3/\text{s}$ of D-T (8-mm pellets and $v \sim 1.5$ km/s). These pellets, however, are not expected to penetrate deep into the ITER plasma.

A recent experiment on ASDEX-U involved the injection of low-velocity pellets from the high-field side of the tokamak. This resulted in significantly improved pellet penetration, high-fuel-coupling efficiency, and unperturbed H mode confinement up to the Greenwald limit. An explanation given is that this takes advantage of favorable drifts that help to transport the deposited pellet material toward the plasma center.

Recently, it was proposed that high-velocity compact toroid (CT) injection may provide the means to deep fuel ITER. The first experiment on deep fueling by CT injection was demonstrated by the Compact Toroid Fueler (CTF) device on the TdeV without any adverse perturbation to the tokamak. A tangential CT injector has begun operation on the STOR-M tokamak at the University of Saskatchewan. The University of California, Davis, group has obtained 1000 consecutive CT pulses at 0.1 Hz, which demonstrates that CT injector technology allows pulsed operation. CT injection experiments on the JFT-2M tokamak are scheduled to begin in November 1997. A second-generation CT fueler (CTF-II) has begun operation on TdeV. A CT fueler is being designed for the Large Helical Device (LHD) stellarator. The University of Washington group proposed a CT-FRC fueler for ITER. A large experiment (TRAP) suitable for fueling JET is in operation, but tokamak CT-FRC experiments are yet to be performed.

Next-Step Devices

While from this summary it might appear that ITER is a paper study, it is important to note that key areas of technology are being addressed by ITER that include the construction of full- and large-scale components of the ITER nuclear component. ITER will use 20 superconducting TF coils and 7 superconducting PF coils plus the central solenoid. The double-walled vacuum vessel is directly attached to the TF coil. Seven critical areas have been identified with ITER, and experimental construction is underway to demonstrate the relevant technology. Two projects are aimed at developing superconducting technology (the central solenoid project by Japan and the TF model coil project by the European Community). Three projects are developing in-vessel components (the vacuum vessel sector project to produce a full-scale sector of the ITER vacuum vessel; the blanket module project aimed at producing and testing full-scale modules of the primary wall, limiter, etc.; and the divertor cassette project to test a full-scale section of the divertor). Two projects will develop the necessary remote-handling technology (the blanket module and the divertor remote-handling projects).

In Italy, the Ignitor project too has experimental work in progress to develop the necessary technology for this small, high-field-ignition tokamak. This device aims to obtain ignited plasmas with high densities ($n_o \sim 10^{21}$ m^{-3}, a parameter demonstrated by Alcator C-Mod) and relatively low temperatures (T_{eo}, $T_{io} \geq 12$ keV). The concept is an extension of the high magnetic field/high-density experiments such as Alcator C-Mod and the Frascati Tokamak (FT) machines in Italy. Supercooled (initial temperature of 30 K obtained using helium gas cooling), normal conducting magnets have been adopted in this design. Detailed engineering design study has been completed. Full-sized prototypes of all the main machine components have been constructed. The construction of the complete central solenoid; the central post; and an entire section ($\frac{1}{12}$th) of the machine, including a fully tiled sector of the plasma chamber, and their assembly have been funded.

The JT-60SU (the Superupgrade, a superconducting device) device is under consideration in Japan. Because high-density steady-state operation is important for a demonstration reactor, a complementary approach is adopted in JT-60SU to have a high Greenwald density limit by choosing $B_T/R \sim 1.1$ to 1.3 T/m as compared with $B_T/R \sim 0.63$ to 0.7 T/m in ITER and $\sim$1.3 T/m in an envisioned steady-state tokamak reactor (SSTR) (the Greenwald scaling, n_e line averaged $\sim B/R$ is unfavorable for large-sized tokamaks at the limited magnetic field of a superconducting magnet).

THE SPHERICAL TOKAMAK

This is a low-aspect-ratio tokamak ($A \sim \leq 1.5$), generally referred to as a spherical tokamak. The low aspect ratio is achieved by minimizing the size of the central ohmic solenoid (possibly removing it entirely) and excluding blankets from this region. In reactor versions, only a single-turn copper TF coil center post may exist. Current drive in large devices must be noninductive, which makes them steady-state devices.

High Beta

The primary advantage is that the Troyon beta is very high. For example, reducing the aspect ratio from 2.5 to 1.2 results in a twentyfold increase in the Troyon beta for the same plasma current. This increases the fusion core power density and results in physically small devices. In a study conducted by General Atomics and described in some detail here, a device with $A = 1.4$, $R = 1$ m (DIII-D size) would produce $\sim$800 MW (thermal), and 160 MW (net electric) and have a gain defined as Q_{plant} = gross electric/recirculating power $\sim$ 1.7. Such a device would have all the operating systems and features of a power plant and would constitute a pilot plant. In this design study, only a single-turn copper TF center post is present on the inboard side. Cooling the center core was not an issue; rather, the minimum physical size of the device was dictated by an acceptable amount of wall loading, which was assumed to be 8 MW/m^2. At this wall loading, the resistivity of the center post will increase by 50% in 7 yr. Replacing the center post does not seem to be an issue, but the high power density is a challenge to the divertor. The presence of a single-turn TF coil also implies the requirement of unusual low-voltage high-current power supplies, which appears to be possible.

Current Drive

In addition to the standard radio-frequency current drive (RFCD) and NB current drive (NBCD) schemes, the study examined direct-current helicity current drive (HCD) and estimated a power requirement of only 10 kW for the assumed broad temperature profiles. However, previous calculations for peaked profiles resulted in powers of 60 to 100 MW, which is twice that for RFCD and NBCD. This large variation in the efficiency needs further study. Experimentally, the Helicity Injected Tokamak (HIT) device at the University of Washington was the first to demonstrate HCD. The HIT-II, now in operation, as well as Current Drive Experiment Upgrade (CDX-U) at Princeton University, the Small Tight Aspect Ratio Tokamak (START) at Culham Laboratory in the United Kingdom, and Free Boundary Experiment (FBX-II) in Japan are expected to examine the HCD concept in greater detail. Operation at high q as is envisioned for a spherical tokamak means that β_p can be sufficiently high to drive significant amounts of bootstrap current. In addition, a large toroidal component of the diamagnetic current is also produced because of the high toroidicity of the spherical tokamak, and calculations show that the total self-driven current can be large.

Spherical tokamak research was first motivated by the very good results on the START device, and even now the bulk of the high-temperature plasma results are from this tokamak. START operates in limiter and double-null configurations. Other observations of START are as follows. It has a vertically stable plasma without any feedback stabilization, and it has no current-terminating disruptions for $A < 1.8$, but it suffers from internal reconnection events (IREs) in which large positive current spikes occur. IREs have also been observed and studied on CDX-U. Also, an NB injector was recently installed. This has led to an increased Murakami parameter ($n_e R/B$) as well as high central betas ($\sim$48%) and high volume-averaged ($\sim$8.5%) betas. Beta is at present limited by the available power; NBI power is being increased, and calculations predict that a beta $\sim$20% may be possible. Energy confinement at low A appears to be as good as or better than most H mode scalings from conventional devices (up to twice ITER-93H). With NBI, there has been no evidence so far for a power degradation, but $P_{\mathrm{NBI}} < 1.5\ P_{\mathrm{OH}}$.

Globus-M is under construction in Russia. The successor to START, which is the Mega Ampere Spherical Tokamak (MAST) ($I_p < 2$ MA), is being built at Culham Laboratory. The U.S. fusion program is considering the construction of a similarly sized device, National Spherical Torus Experiment (NSTX) at Princeton or University Spherical Tokamak Experiment (USTX) at the University of Texas.

STELLARATORS AND TORSATRONS/HELIOTRONS

These can provide a currentless steady-state MHD equilibrium configuration of plasma using only the external fields produced by the windings outside the plasma. These have the advantage of having no current disruptions. The rotational transform that is needed to confine the plasma is formed by the external coils. A stellarator has TF coils plus l pairs of helical windings. The helical windings next to each other carry current in opposite directions. In advanced devices such as Wendelstein VII-X (WVII-X) (approved for construction in Germany), twisted (but a modular set) TF coils produce all of the necessary magnetic fields. In addition, there are no toroidal windings making coil and blanket replacement easier. A torsatron/heliotron has only l helical windings (instead of $2l$) with currents all in the same direction. The LHD (construction >75% completed) in Japan is an example of this. The helical windings also produce the toroidal fields, so there are no TF coils. For heliotrons/torsatrons, vertical field coils are needed to counteract the vertical field of the helical windings. In a heliotron/torsatron–type configuration, a built-in-separatrix configuration at the free space between helical coils leads to a helical divertor structure. For the stellarator, a local island divertor concept has been developed and studied in WVII-AS. Such a divertor would be used on WVII-X. Such localized divertors would be used during the initial operation of the LHD. Preliminary experiments to demonstrate this concept were performed on the Compact Helical System (CHS) (Japan).

An international helical system database on τ_E has been compiled [the International Stellarator Scaling (ISS 95)]. An interesting result is that the L-mode confinement of the stellarator and tokamaks is similar. This would mean that the gross confinement properties should not be dominated by the parameters, which are very different in the two concepts, i.e., magnetic shear, major rational values of the rotational transform, and plasma current. A difference in confinement is that there exists evidence for pinches in the particle and possibly energy transport areas in tokamaks, whereas in the stellarator no pinches have been seen. In an experiment in which up to 30-kA current was driven in WVII-AS, the density profile remained flat. In comparison, the ASDEX data (a similarly sized tokamak) showed a peaked density profile. However, the Ware pinch velocity at 30 kA is only 3 cm/s, whereas it is 25 cm/s for the ASDEX discharge. To study whether higher toroidal electric fields are needed to create a pinch, experiments at higher collisionality are being conducted.

The density dependence indicated by the ISS scaling is absent in modern tokamak scalings that include auxiliary heating. The density limit is much higher in a helical device than in a tokamak of similar size. A possible reason is that helical devices do not have current disruptions. The maximum plasma parameters obtained to date are as follows: $T_e(0) = 3$ keV, $T_i(0) = 1.6$ keV, $\langle n_e \rangle$ 2 to 3×10^{20} m^{-3}, $\langle \beta \rangle = 2.1\%$, and $\tau_E =$ 40 ms (not all simultaneously). For next-generation devices, β of 5% is projected.

Improved confinement modes have been obtained. The first is the high T_i mode, which has been observed in NB-heated plasmas in Heliotron-E (H-E) (Japan) and WVII-AS at low densities. The value τ_E is ~40% larger than in the L mode (in H-E). Both T_i and n_e are peaked. The second is the pellet mode, where in H-E, pellet injection caused the T_i profile to become peaked and $T_i(0)$ increased after pellet injection. The third is the H mode, which was observed in WVII-AS and CHS. The features are similar to those in tokamaks. The fourth is the Reheat mode; in CHS, an increase in the stored energy was observed together with density peaking when gas puffing was turned off in a high-density regime. The peaked n_e profile is triggered by the decrease in the edge neutral density.

In the stellarator, the radial profile of the electron thermal diffusivity is relatively flat, or it even increases toward the center (in contrast to tokamaks where it sharply decreases toward the center). In the L mode, χ_e and χ_i are similar in magnitude and profile. In improved modes, χ_i decreases toward the plasma center (~0.5 to 1 m^2/s); the values are close to those observed in improved modes in tokamaks.

The Shafranov shift was observed to be reduced in WVII-AS, as expected from a reduction in the Pfirsch-Schlüter current. This was a confirmation of a theoretical result that was an objective of configuration optimization in WVII-AS.

LHD expects its first plasma in 1998, while WVII-X is scheduled for operation in 2004. Extrapolation of the WVII-X concept (the Helias reactor design at the Institute for Plasma Physics, Garching, Germany) to a reactor results in the following parameters: $R = 22$ m, $a = 1.8$ m, and B on axis $= 5$ T with an output power of 2.5 to 3.8 GW at a plasma volume ≤ 1400 m^3.

REVERSED-FIELD PINCH

This is a toroidal device similar to a tokamak. Differences are that B_θ is similar in magnitude to B_ϕ and q is always <1. RFPs are stabilized by large magnetic shear. Toroidal plasma currents produce B_θ. The toroidal field is partly produced by a weak TF coil but mainly by large poloidal currents flowing in the plasma. The other difference is that the toroidal magnetic field on the axis reverses itself from the imposed direction and continues up to the wall. This field reversal is generated by the plasma in a process known as the dynamo effect as a result of relaxation processes.

This dynamo effect is not fully understood. Research continues to be directed at understanding the mechanisms responsible for sustaining the RFP magnetic configuration and transport in these plasmas. On the TPE-1RM20 device (Japan), a new dynamo mechanism (diamagnetic dynamo effect) was found, which indicates that pressure gradients may play a role in the dynamo activity. A new operation mode was found and named the improved high-theta mode, in which the values of the electron density, poloidal beta, and τ_E were doubled.

On the Madison Symmetric Torus (MST) device (United States), by application of a transient poloidal inductive electric field, a fivefold enhancement of τ_E was obtained as a result of reducing the amplitude of the tearing-mode fluctuations. In addition, methods are being developed for current profile control, which are expected to reduce transport associated with current-driven fluctuations.

On the Extrap T2 device (Sweden), resistive shell operation and confinement was investigated (in RFPs, radial equilibrium control by external fields requires a resistive wall, while the RFP equilibrium requires high wall conductivity to stabilize MHD modes). Wall boronization and helium-glow discharge cleaning has resulted in much better plasma performance (compared to that 2 yr ago). Pulse lengths about eight times the magnetic field penetration time were obtained. On the Separatrix Test Experiment (STE-2) device (Japan), the influence of external helical perturbations on RFP dynamics was investigated, and it was found that the mode-coupling process could be controlled by external helical fields.

FRC, SPHEROMAK, AND TANDEM MIRROR

The FRC is a high-beta CT plasmoid containing only a poloidal magnetic field. On the FRC Injection Experiment (FIX) device (Japan), a rapidly rising magnetic pulse from a pair of half-turn coils was used to heat a translating FRC apparently without confinement degradation. The spheromak CT contains toroidal and poloidal fields of similar magnitude. In an experiment (Japan and the United States) in which two spheromaks with opposing toroidal fields were merged together in the axial direction, a high-beta FRC resulted with increased ion temperature (due to magnetic field annihilation). A tandem mirror has a cylindrical magnetic configuration with complex magnetic plugs at the ends to reduce losses. On the Gamma-10 device (Japan), direct electron heating in the central cell reduced the energy drain from the hot ions due to reduced electron drag and resulted in more effective ion heating in a hot ion mode.

INERTIAL CONFINEMENT FUSION

ICF involves compressing a target pellet to ignition by powerful laser beams or ion beams. Ablation of the outer shell of the target causes a spherical shock wave to be launched inward that can compress and heat the fuel to ignition. The resulting small thermonuclear explosion deposits energy on the chamber walls and surrounding blanket. In general, the drivers (lasers or ion beam) must focus ~1 to 10 MJ of energy in ~10 ns.

For high-energy gains, the compressed value of the final fuel is $\sim$1000 g/cm^3. In implosion experiments at GEKKO XII (Japan), high-energy compression up to 600 g/cm^3 has been demonstrated. The energy required for such compression is much less than the energy that would be required to heat the entire target up to the ignition temperature.

Beam Interaction with Plasma

When the laser strikes the target surface, some of the surface atoms will be vaporized and ionized, forming a plasma around the target. The laser beam energy will be partly reflected by this plasma and partly absorbed. Two processes, stimulated Brillouin scattering (SBS), which backscatters the incident light, and stimulated Raman scattering, which preheats the fuel by superthermal electrons, are detrimental for beam absorption. When a prepulse precedes the main laser pulse, the early plasma formation enhances SBS during the main laser pulse, reducing absorption. Preheating raises the internal fuel pressure and prevents the attainment of high fuel densities. Thus, it is desirable to delay heating until compression is nearly complete.

In ignition-sized capsules, the laser beam needs to penetrate a significant distance inside the surface plasma, so that the target can be sufficiently heated. Increasing the laser frequency allows deeper penetration of the beam because the critical density for reflection is higher. This aspect as well as the demonstrated high-conversion efficiency of the KrF laser beam (laser energy/stored energy) of $\sim$10%, and the capability to operate this laser system in a repeated mode, makes this best suited for laser fusion. However, the advanced state of glass laser technology and capability of tripling the frequency of the neodymium-glass laser (which puts the final frequency close to the KrF laser frequency) and the near-term goal of demonstrating ignition on a single-pulse device has resulted in the large facilities using this system.

Filamentation and Target Ripple

Self-focusing of a laser beam in a plasma may cause the laser beam to split into several smaller self-focusing beams; the process is called filamentation. Filamentation causes nonuniform irradiation of the target, which can cause the Rayleigh-Taylor instability. Rayleigh-Taylor instability can also result from surface ripples on the target surface. Pellet asymmetry problems are expected to be more severe at short wavelengths. Several developments have occurred during the past several years to overcome this problem.

Indirect Drive

In this scheme, the pellet is contained in a cavity (a hohlraum) fabricated out of a high-Z material. The beams are directed onto the cavity surfaces, which results in the production of X rays that heat the fuel in a symmetric manner without the original problems of filamentation. A negative aspect is that the coupling of the beam energy to the target is reduced (only $\sim$10 to 15% in present experiments and anticipated improvements up to 25% after optimization). Gas-filled hohlraum targets are proposed to achieve ignition at the National Ignition Facility (NIF). Experiments are now being performed with the Nova laser (United States) using methane- and propane-filled hohlraum targets to assess the effect of gas fill on radiation drive and symmetry and to verify modeling of NIF gas-filled hohlraums. Now under construction, NIF is expected to produce the first controlled ignition within the next 10 yr. The $\sim$9.5-mm-long, 5.5-mm-diam NIF hohlraum target contains a 1.1-mm-radius target pellet. The target is heated with a 1.8-MJ laser pulse and is expected to produce between 1 to 10 MJ of fusion energy. The hohlraum scheme is now used by many facilities.

Improved Targets

Among other schemes, this involves depositing a layer of foam on the fuel pellet, which diffuses the effect of nonuniform laser imprinting due to filamentation. In addition, modern targets also include a coating of high-Z material over the foam. The incident laser beam converts the high-Z material into a strongly radiating plasma, which uniformly ionizes the foam by soft X rays. Although driven directly, the presence of a high-Z shell and the indirect coupling of the laser energy to the pellet (by X rays) has some resemblance to the hohlraum concept. Variations of this concept are being developed at General Atomics (United States), Institute of Laser Engineering and Institute for Laser Technology (Osaka, Japan), Commissariat à l'Energie Atomique (CEA) (France), and other facilities.

Beam Smoothing

Direct drive has the advantage of high-beam coupling to the target (>50% possible). Hence, several schemes are being developed for beam smoothing that will be necessary for a uniform spherical compression. For example, one scheme involves the use of a random phase plate to introduce a randomness in the phases of the beam. This effectively creates a smoother beam by overlapping many small beamlets of random phase. Variations of this and other schemes are used at the Gekko VII facility to reduce the nonuniformity root mean square (rms) to <10%, which has resulted in a 50% reduction of the growth rate of the Rayleigh-Taylor instability. Near-term plans are to heat a 600 g/cm^3 compressed target using the 13th arm (to be installed) of the Gekko XII system. The 13th arm will deliver a short pulse (100 J in 1 picosecond) and may demonstrate Fast Ignition. In fast ignition, the compressed fuel with a high density created by a conventional laser implosion is heated by an ultraintense short-pulse laser (a few picoseconds) before the core disassembles (less than a few picoseconds). The concept may reduce the required laser energy for ignition because only a small part of the pellet core needs to be ignited. The burn wave may ignite most of the remaining fuel.

Beam smoothing on the Omega system (United States) has resulted in the attainment of the highest thermonuclear yield (1.4×10^{14} D-T neutrons) and yield (>1% of scientific break-even) ever attained in laser fusion experiments.

Smoothing of the Nike KrF laser beams (United States) has resulted in each of the beams having a nonuniformity of $\sim$1% rms in a 4-ns pulse. This beam nonuniformity is a factor of $\sim$10 better than any existing glass laser when used in the ultraviolet. With Nike, multiple beams (37 to 44) are overlapped on the target, reducing this nonuniformity still further to a few tenths of a percent. Preliminary experiments were very encouraging. Using a smooth foil target resulted in mass nonuniformity below the diagnostic measurement level.

Several smaller laser systems are in operation around the world to study the physics issues involved. At the Imperial college in the United Kingdom, foam-buffered targets are studied. Development of high-yield capsules for NIF is proceeding at Los Alamos National Laboratory. At the ASTERIX laser facility in Germany, the extent of allowed preheat for various high-Z ablator material/hohlraum geometry is being studied. A labyrinth hohlraum, suited for single-beam laser systems, in which a cone stops the incoming laser pulse such that the primary X rays cannot reach the sample, has been developed. At the Electrotechnical Laboratory in Japan, a KrF system "Super Ashura" is being developed to establish KrF laser driver technology and to perform experiments at a 5-kJ level. In addition to one-dimensional simulations of a megajoule ablator, development of a megajoule laser system is underway at CEA-Limeil-Valenton in France. The Plex Corporation and Pulse Sciences (United States) have proposed a KrF-based fusion facility, while the Osaka group has proposed a diode-pumped solid-state laser fusion power plant.

Light-Ion Beams

Compared to lasers, light-ion beams have the following advantage: ~25% beam efficiency, which means that lower-gain targets could be used. Therefore, less stored energy is required (10 MJ compared with 50 to 10 MJ for lasers). Light-ion beams may produce less target preheat. The Particle Beam Fusion Accelerator II (PBFA II) at Sandia National Laboratories (SNL) has the world's largest light-ion-beam accelerator. This is used to indirectly drive capsules by radiation that is produced when the high-energy lithium beam travels through a high-Z radiation case. The radiation is deposited in a low-density foam that surrounds the fusion capsule. An issue for ion beams is the transport of beams over long distances (this is called standoff). A promising concept is self-pinched transport. In this scheme, the ion beam is focused down to a small radius, stripped, and then injected into a low-pressure gas. The azimuthal magnetic field produced by the ion current confines the beam. An issue is whether gas breakdown will produce sufficient current to allow self-pinched transport. Experiments to demonstrate this are being initiated on GAMBLE II (1- to 1.15-MeV protons at 150 kA and a diode outer radius of ~10 cm) at the Naval Research Laboratory and on SABRE (3.5- to 6-MeV lithium beams, 150 kA, and a diode radius of ~6 cm) at SNL.

Heavy-Ion Beams

These have the advantages of even more improved coupling of the beam energy to the target, low-beam current (~10 kA), stiffness of heavy beams that eases focusing, high-beam conversion efficiency (~30%), and ability for multipulse operation with present technology. Recent calculations (EURATOM-ENEA in Italy in collaboration with Germany and Spain) show that thermonuclear fusion can be attained with a beam energy of ~2 MJ. The reference target is a two-sided irradiated cylindrical hohlraum, enclosing a spherical fusion target. Lawrence Berkeley Laboratory and Lawrence Livermore National Laboratory have constructed a 2-MV, 800-mA potassium injector. Because reactor applications require ~10-kA beams at 10 GeV, heavy-ion-beam fusion is the least developed of the various inertial fusion concepts.

Results presented at this conference show that significant progress in fusion physics has been made (discovery of several advanced confinement regimes and D-T operation), which is further demonstrated by the action of countries that have approved the construction of large next-generation fusion facilities. These are the NIF laser fusion project in the United States, the WVII-X stellarator in Germany, and the LHD in Japan. In addition, other projects are awaiting full approval. These are the Ignitor device in Italy, the NSTX device in the United States, the CEA-LV laser facility in France, and JT-60SU in Japan. Construction and operation of the ITER device will help to address the fusion engineering issues related to a reactor.

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