Fusion, Magnetic Confinement—Addendum

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INTRODUCTION

In the 1990s there has been significant progress in the development of the tokamak concept and serious study by an international research group [the International Thermonuclear Experimental Reactor (ITER) team] on the feasibility and design of an experiment to demonstrate fusion power production. In this addendum we present an overview of the status of tokamak research in relation to achieving this goal.

1. TOKAMAK REACTOR CONFIGURATION

1.1 Magnetic Field

Recall that a tokamak is a toroidally shaped plasma in a magnetic field as shown in the schematic diagram of Fig. 1. Its cross section is shown in Fig. 2, with typical reactor dimensions. The dominant magnetic field is in the toroidal direction and is established by toroidal field (TF) coils that poloidally loop the plasma. A smaller, poloidally directed magnetic field arises from toroidally directed current flowing in the plasma itself and toroidal currents flowing in poloidal field coils that are outside and parallel to the plasma (currents external to the plasma control the shape and position of the plasma). All these currents produce magnetic field lines that circulate in the toroidal and poloidal directions and form magnetic flux surfaces. The q-value (a parameter that will be frequently referred to) of a surface is the number of times a field line circulates in the toroidal direction in one poloidal circuit. The surfaces are represented by closed surfaces in Fig. 2.

1.2 Central Solenoid

The central solenoid is used to supply the toroidal voltage that sustains the plasma current. The voltage arises from the time rate of change of the entrained magnetic flux in the coil of the central solenoid. However without other current drive methods being imposed, the establishment of the plasma equilibrium and the sustainment time of the discharge is limited by the time integral of the emf that is obtained by reversing this magnetic flux. This time is expected to last about 1000 s in a fusion producing plasma.

1.3 Fusion Burn

Charged particles are trapped in continuous spirals that closely follow the entrapped field lines. This containment allows the particles to be heated to temperatures where fusion reactions occur. For a deuterium-tritium mixture, this temperature is in the range 10-25 kev where we use temperature in units of energy (where $1\,\mathrm{ev} = 1.16 \times 10^4\,\mathrm{K}$). When deuterium and tritium fuse ($d + t \Rightarrow {}^4\mathrm{He} + n + 17.6\,\mathrm{MeV}$), the 14 MeV neutrons leave the plasma and are absorbed by the walls. Their energy is to be used to produce steam that drives an electrical power plant. In addition the neutrons are needed to breed tritium (tritium is radioactive with a half-life of 12 years and is available only in limited amounts from fission reactors). Hence it will be necessary to surround a fusion reactor with a lithium blanket (see Fusion Technologies) to breed tritium [$n + {}^6\mathrm{Li} \Rightarrow t + {}^4\mathrm{He} + 4.8\,\mathrm{MeV}$; $n + {}^7\mathrm{Li} \Rightarrow t + {}^4\mathrm{He} + n' - 2.5\,\mathrm{MeV}$]. The 3.5 MeV charged alpha particles (${}^4\mathrm{He}$) should be retained in the tokamak and their energy transferred to the background plasma by collisions. Then, even without external heating, a “burn” is established, whereby the plasma fuel remains hot while losing energy due to bremsstrahlung, atomic radiation and convection of heat to the “cold” solid walls. Ignition occurs when the fusion produced alpha particles supply the required heating instead of an external source.

1.4 Divertor

Figure 2 also illustrates a single divertor configuration where there is a special magnetic flux surface called the separatrix (the one where the poloidal field lines appear to cross). Field lines beyond the separatrix (the scrape-off layer) impinge on structure called divertor plates designed to handle large escaping heat loads (charged particles that diffuse outside the separatrix impact here). Gas formed from the recombined plasma is to be pumped away from the surrounding divertor chamber into a closed tritium recovery system.

2. OPERATION OF A BURNING PLASMA

2.1 Lawson Condition

In order for fusion reactions to be thus sustained, the confinement properties at fusion temperatures must satisfy the Lawson criterion, $n\tau_E \approx 2 \times 10^{20}$ m^3 s (here n is the d-t ion density and τ_E the overall energy lifetime). The energy lifetime is determined by transport arising from particle collisions and plasma induced short wavelength turbulence. At fusion temperatures, the density is limited by magnetohydrodydamic (MHD) stability. This sets a limit on the parameter beta, the ratio of the average pressure (the particle density times the temperature) to magnetic field pressure. In standard tokamak operation, where the current density is largest in the plasma's center, there is a further MHD limitation that prevents the q-value (which is inversely proportional to the current density), from being too much less than unity. In the standard case the beta is limited to about 5%. In addition there exists a less well understood empirical density limitation, perhaps associated with charge particle recombination, known as the Greenwald limit, which seems to limit the maximum density to a level proportional to the mean current density of the discharge.

2.2 Magnetic Field Limits

Another limitation is set by the magnitude of the magnetic field that can be imposed. With pulsed operation ($\sim 10\,\mathrm{s}$), TF coils with about a 20 T peak field (producing a 13 T field at the plasma center), have been designed. However, superconducting magnets are generally considered to be needed for a fusion experiment designed to operate in quasi-steady conditions of order 1000 s. Then peak fields are limited to 12 T, which allows a 6 T field at the plasma center.

2.3 Empirical Scaling

The energy lifetime of typical tokamak discharge is uncertain as it is primarily governed by incompletely understood turbulent processes. Thus the extrapolation of tokamak parameters to larger size reactor regimes is typically based on empirical scaling to the experimental database, obtained from many different installations. A reasonable fit to many experiments is achieved by assuming that the energy lifetime is proportional to algebraic powers of the physical parameters defining the experiment (such as plasma current, magnetic field, plasma heating rate, etc.), with the value of each power the same for all the experiments. Some general conclusions from this procedure is that confinement improves increasing plasma current and increasing toroidal magnetic field, but degrades with heating power. When this law is applied to an ignition experiment, the extrapolated lifetime is compatible with achieving ignition, but with some uncertainty due to the variance in the data.

2.4 Heat Load

The dimensions shown in Fig. 2 arises as result of conforming to the constraints mentioned above. They lead to a total fusion power production in the gigawatt regime (D.E. Post and N.A. Uckan, 1992). One fifth of this power (from alpha particles) directly heats the plasma. The heat exhausts on plasma facing material with a rather modest average power load of about 0.2 Mw m^{-2}. However, should most of this heat impinge on the divertor plate, an unacceptable 40 Mw m^{-2} heat load would arise.

Operation with a divertor is viewed as essential. It has been found to be extremely important to control the impurities that enter the plasma, to design efficient pumping methods, and to establish the conditions for optimum plasma confinement in the so-called H-mode. The H-mode of operation improves the energy confinement time a factor of 2 or more compared to an alternate mode of operation, the so-called L-mode (see below for further discussion).

A fix to this heat exhaust problem is to radiate away most of the exhaust energy. For example if 1500 Gw of fusion power is being produced, then the plasma is directly heated by the 300 Mw in alpha particle production.

Roughly 100 Mw radiates as bremsstrahlung. Impurity seeding at the plasma edge and diverter throats is required to radiate away about 150 Mw so that the remaining 50 Mw of heat flux in the escaping plasma allows a tolerable power density load of less than 10 Mw m^{-2} at the divertor plates. Favorable results are now available in several tokamak experiments that demonstrate that the heat load on the divertor plates is indeed reduced by such impurity radiation shielding, in a so-called semi-detached mode of operation, while maintaining relatively good purity of the plasma. An important associated issue is whether optimal H-mode confinement properties can be achieved in semi-detached operation. Most experiments observe confinement degradation, but recently there has been results in several tokamaks demonstrating that in semi-detached operation confinement properties close to the typical H-mode level is achieved.

2.5 Pumping

Pumping in the divertor region is extremely important in reactor operation. The helium accumulating in the core must be pumped away. Otherwise the burn quenches after ignition is achieved, as helium ash is replacing plasma fuel. Experiments have shown that the ratio of helium to energy lifetime is about 5, and with pumping and d-t gas puffing the helium accumulation can be limited to 10% of the plasma density, an acceptable level. The pumping in the divertor system is a principal control of the plasma impurities as well. The density of seeded impurities need to be peaked near the scrape-off layer but diffuse in the plasma bulk. Present experiments are giving positive indications that the pumping control of impurities needed in a reactor experiment can be met.

2.6 Ignition Physics

There are several intrinsic physics issues associated with achieving ignition and burn. One is the control of the parameters of the plasma. There can be an ignition induced instability, where once the fusion burn dominates, the plasma heating rapidly increases the plasma pressure. The increased pressure can cause the system to enter unstable MHD regimes where confinement deteriorates or is even lost. The conditions for such excursions are believed to be understood and in principle can be avoided. Ultimately the appropriate operation must be experimentally demonstrated.

Another reactor issue is the physics associated with alpha particle confinement. In a reactor the alpha energy is to be absorbed by the plasma. It is possible for energetic alpha particles to diffuse to the plasma edge and then embed themselves into the plasma facing material. It is estimated that a limit of only 5% of the generated energetic alpha particles can be lost in this way if serious wall deterioration is to be averted.

Energetic alpha particles near the edge are primarily lost due to complicated motion arising from the presence of toroidal magnetic field ripple resulting from the spacing of a finite number of TF coils in the machine. It has been shown that if only the energetic alpha particles born near the edge are lost, there is a tolerable loss with a reasonable number of TF coils. However there is still a concern that alpha particles can be fed into the edge region from plasma instabilities. One possibility is that Alfvén wave stabilities can be spontaneously generated with the alpha particles themselves the instability source. Another possibility is that alpha particles are lost due to MHD activity of the plasma, which either directly drives the alpha particles into the loss region or indirectly triggers the Alfvén instability by causing a rearrangement of the alpha particle pressure profile. Such scenarios, with loss, have been observed in experiments that use energetic particles arising from neutral beam injection or ion-cyclotron rf heating. In the TFTR tokamak, where up to 10 Mw of fusion power has been produced, the alpha particle driven Alfvén wave has been observed, but in these cases observed saturation level is too low to cause significant energetic particle loss. The theory for the Alfvén wave instability and loss is still not sufficiently well developed for quantitative predictions to reactor cases, but it is likely that the Alfvén instability will prevent some otherwise desirable set of plasma parameters from being achieved.

3. PLASMA CONFINEMENT ISSUES

3.1 H-mode

Clearly energy confinement time is a critical parameter in determining the feasibility and design of fusion power production. The plasma properties can be significantly altered if turbulence conditions are changed. This appears to be the case when there is a transition from L-mode to H-mode operation. In H-mode a pedestal in the temperature appears at the edge of the plasma with a radial thickness typically of an ion orbit width. In this layer there is a flowing plasma with radial shear. A widely accepted explanation of why H-mode is established (K. Itoh and S. Itoh, 1996), is that the escaping plasma causes a torque to be generated that drives a shearing flow. The shear in turn suppresses fluctuations in the pedestal region by, either "ripping" apart the coherence of the fluctuations so that they saturate at a low level, or preventing unstable fluctuations from appearing. It is experimentally observed that a threshold power level is needed to establish an H-mode and this observation is consistent with the hypothesis for the cause of shear flow. The determination of this level is still uncertain; there is no generally accepted quantitative theory and empirical studies have too much scatter to predict whether, in the reactor operation described above, there is enough power flowing through the edge to establish an H-mode. This is an important issue that is to be resolved in ongoing studies.

3.2 Transport Barriers

The studies of the H-mode have been crucial in the discovery of new ways to improve energy confinement (K. Burrell, 1997). It has been found in many tokamaks that by applying torques in either narrow or wide regions of the plasma, by use of either neutral beams or rf sources, transport barriers can be established in the core of the plasma. High temperatures have been established in the central region, with a large drop in the temperature taking place in the transport barrier. It has been further found that establishing this barrier is aided by having a hollow current profile, which leads to a reversed q-profile that has a local maximum at the plasma center and a minimum q-value, typically ~ 2, further out radially. The thermal barrier is generally established near the minimum q-position. Stability theory indicates that in the region where q is radially decreasing, the strength of the diamagnetic drift-magnetic curvature driven instabilities [see Fusion, Magnetic Confinement] are reduced, and the flow shear that is established is then enough to stabilize the modes completely. Some experiments report that the residual transport can be explained by transport due only to collisional effects.

The major issue with modes that have hollow current profiles, is that it is transient, unless a tailored emf can be applied from auxiliary current drive methods involving rf or neutral beam sources. It needs to be demonstrated that power requirements for such a drive are compatible with reactor operation. Another benefit from the hollow current profiles is that they lead to plasmas where a large fraction of the current is in the form of bootstrap current [see Fusion, Magnetic Confinement], which does not decay in absence of voltage. If this can indeed be established, then true steady state operation in a reactor with non-ohmic current drive power is feasible as it leads to acceptable power requirements for driving the non-bootstrap current component. Steady state operation is extremely desirable. It avoids what may be delicate plasma control issues associated with start-up and shutdown, as well as thermal stresses associated with repetitive operation that lead to material deterioration.

3.3 Transport Theory

There has been considerable development in theory to achieve transport predictions based on plasma physics principles. The theories used are still in a post-processing phase, i.e. they explain the temperatures achieved in a large number of experiments, but have not predicted the temperature of any experiment before it was performed. One reason is that a theory for energy transport is easier to develop than a theory for particle transport. Hence to determine temperature, a density profile has to be taken from experiment. In addition the transport processes in the core of the plasma are thought to differ from the processes at the edge, and typically the various predictive methods make allowances for less understood edge transport processes by either using experimentally observed temperatures at the edge or by adjusting edge transport coefficients to match empirical observation. As a result many theory based transport calculations produce good correlation of the predicted and observed energy lifetimes for a variety of experiments (Connor, *et al.* 1997). However, when the theories are extrapolated to reactor type plasmas, significant deviations are predicted. One theory predicts that gigawatt fusion power operation can be expected only if a high edge temperature, 4 kev at the inner edge of the pedestal can be achieved, while the another theory indicates that such fusion power levels can be achieved even with a modest edge temperature. In such case the former theory would predict a fusion power output that is about a factor of 2 larger than the external heating input which would give a few hundred megawatts of fusion power when about 100 Mw input power is available.

The resolution of the issue of attaining a predictive theory with a high degree of confidence is of high priority. In the near future there will be critical tests to resolve existing discrepancies. Nonetheless, it is clear that there has been a great deal of basic progress in the understanding of the turbulent transport problem and we can expect that the theoretical reliability of this approach to improve rapidly.

3.4 MHD Considerations

If the plasma's MHD stability characteristics can be improved it can lead to better reactor properties. At a given temperature, a greater density can be achieved, making it easier to satisfy the Lawson criterion. One of the important parameters controlling MHD stability is the plasma shape. Note the vertical elongation and the triangular shape of the outer flux surfaces in Fig. 2. The decreasing pressure profile at the pedestal is a destabilizing effect on the outer part of the surface and is a stabilizing effect on the inner part of the surface. The triangularity allows stronger weighting from the stabilizing inside than would arise without this type of shaping. Then the temperature drop at the pedestal where the H-mode formed can be increased to a limit apparently determined by MHD oscillations known as elms. As one example we note that in 1996 the best bulk confinement parameter $n\tau_E T = 1.5 \times 10^{21}$ m^{-3} s kev was achieved in the JT-60U experiment, in part due to shape optimization.

There are many experiments that achieve higher beta values, but they frequently have MHD fluctuations that reduce the energy confinement time or cause a disruption where the plasma discharge rapidly disintegrates. One route to a disruption is through a so-called locked mode, where a perturbed wave "attaches" to an imperfection of the stationary magnetic field, which can cause deterioration of the discharge. Locked modes are avoided with a sufficiently large plasma rotation and hence enough rotation needs to be generated in a reactor experiment.

There is a great deal of optimism that discharges with reversed q-profiles in the center can attain significantly increased beta values, particularly when a close fitting conducting wall is used. However, in this case slow growing modes still arise due to the finite wall resistivity. Such modes can in principle be controlled by feedback with low power sensing circuitry. More experiments on this topic are needed and appropriate reactor designs still need to be developed.

3.5 Reliability

The demonstration of reliability is another challenge for tokamak operation. In conventional operation, pulsations, such as sawtooth oscillations and elms, are common. Typically the sawteeth arise when the q-value of the discharge is somewhat below unity and the pressure profiles peaks too strongly within the $q = 1$ surface. The time interval between sawtooth pulses is typically a fraction of the overall current diffusion time (the time interval is of order 100 ms in the larger present day machines and is expected to be about 1 s in a tokamak reactor). The pulsation flattens the pressure profile within the $q = 1$ surface, but it also has a beneficial effect of preventing further steepening of the central current profile, a destabilizing tendency. These pulsations generally cause rather small perturbations at the edge so that ordinary sawteeth by themselves are not likely to induce heat flow problems at the divertor plates. However, in some operation energetic particles induce a change of plasma behavior that cause the sawtooth oscillations to be suppressed for long time intervals. Unfortunately, at times, a giant sawtooth pulse appears that disturbs edge conditions and can even disrupt the entire plasma discharge. Such behavior must be controlled in reactor operation.

The elms, mentioned earlier, are a plasma edge perturbation which cause enhanced heat pulses to outflow at regular time intervals. There is an acceptable type of elm operation, so-called grassy elms, where the pulses are frequent but low level, which lead to heat loads that are readily handled, good global confinement characteristics, and an observed advantageous property of keeping impurities from accumulating in the main part of the plasma. However, elms arise that give larger heat pulses and less favorable plasma confinement. A further difficulty is that a large elm pulse can cause termination of H-mode operation which will then lead to a significant reduction of the fusion output unless active counter measures to restore H-mode operation are found.

Large disruptions are known to arise in tokamaks which lead to the loss of the plasma discharge. Frequently these disruptions are uncontrolled and are more likely to arise for tokamaks that operate at the limits of acceptable operation. Of concern is that large disruptions can cause damage to plasma facing wall components due to localized heat deposition, the formation of so-called halo currents where the plasma current penetrates the surrounding conducting wall and the formation and subsequent wall bombardment of relativistic "run-away" electrons accelerated from the inductive electric field.

Studies are in progress to develop reliable ways that disruptions can be guided to a "soft" landing, where the heat is distributed benignly over the entire surface of plasma facing walls (for example, from injection of so-called killer pellets, that cause the heat to radiate), and the current in the remaining cold plasma, together with its associated magnetic field energy, is allowed to dissipate on the time scale of a second, a time interval long enough so that the inductively produced electric field is too low to cause run-away electrons.

4. CONCLUSIONS

4.1 Alternative Tokamak Approaches

On the basis of the knowledge known today there is a high confidence level that a fusion power producing experiment can be designed to operate, at a minimum, in a fusion power amplification mode where several hundred megawatts of power are produced. A more challenging goal is to reliably achieve quasi-steady burn conditions or a high enough energy amplification level that produces gigawatt power levels. Such success requires maintaining optimal tokamak performance (such as H-mode operation, steady alpha particle heating, limited parameter excursions from pulsations, etc.) in the standard type discharge discussed above.

Alternative directions of research will proceed either in conjunction with, or as an alternative to, the direct implementation of a fusion power experiment. Perhaps the most exciting development in the mid-1990s is the demonstration that turbulence levels can be dramatically diminished, leading to enhanced confinement, as a result of the combination of plasma flow shear and establishing q-profiles that decrease radially over much of the discharge. The future challenge in the operation of this type of discharge is to demonstrate that such conditions can be steadily maintained with modest external power control that extrapolates properly to reactor size plasmas. The intrinsic boot-strap current in such discharges is expected to be high enough so that the power needed to maintain true steady state reactor operation can be achieved. Significantly, non-inductive current drive experiments in the Tore-Supra and in other tokamaks have already proven that plasma discharges can be sustained indefinitely, but the needed efficiency is still to be demonstrated.

Future research will also focus on achieving higher plasma pressure operation by implementing conducting wall stabilization with feedback to control slow growing resistive wall instabilities. The new methods of obtaining enhanced confinement may also lead to alternative designs for plasma shaping. Other studies are examining whether plasma confinement properties can be improved by increasing a tokamak's minor to major radius.

There is extensive interest in building an experiment demonstrating ignition and a short burn, but which does not address long time reliability issues (B. Coppi, 1994). By using the highest possible magnetic fields, plasmas can be established that are well within intrinsic parameter limits. Hence in such an experiment the likelihood of achieving a proof-of principle controlled burn is high.

Another alternative is to attempt to obtain net fusion power operation when the d-t temperature is greater than the electron temperature. It is significant that such operation characterizes experiments that have achieved some of the highest Lawson parameters in JT-60U, JET and TFTR. If the $n\tau_E$ parameter can be increased another factor of 3-4 in the former two machines, the condition of a relatively compact reactor operation would nearly be established. There is a problem with maintaining this mode, because as confinement improves the electron and ion temperatures tend to approach each other, and in the process degrade the favorable features of this type of confinement. A novel mechanism, known as energy channelling has been suggested, where through the controlled stimulation of rf waves, the alpha particle energy directly heats the d-t fuel rather than the alpha particles primarily heating the electrons (as generally occurs). The technical feasibility of this idea is still under study.

4.2 Other Concepts

The most developed alternative to tokamak research is the stellarator concept (see Confinement, Magnetic Fusion), where charged particles are confined by the three dimensional magnetic field established by currents in external coils. This concept is intrinsically steady state, some plasma confinement regimes have been shown to be comparable to similar tokamak confinement regimes, and experiments have been free of disruptions. Of principal concern is that the achieved beta values in stellarators is generally lower than in tokamaks and in most designs a large class of energetic particles are not contained in the magnetic fields. A positive recent development has been the design of so-called quasi-helical fields, which does indeed produce confinement of energetic particles. However, this design still needs to be made compatible with the achievement of reasonably large beta limits. Another advantage of a stellarator is its typical property of a radially increasing q-profile. This may allow easier access into regimes demonstrating enhanced confinement.

A new large stellarator in Japan, the LHD, will be in operating in 1997, and it will further test the prospects for achieving high grade plasma confinement suitable to reactor operation.

Research in pinches still continue with the hope that they can be made into compact fusion sources. Though these experiments are far from achieving their fusion goals, they often have intrinsic scientific interest. They have been used as x-ray sources and for such basics physics studies as the understanding of the interaction of plasmas with solids, plasmas with lasers and in addressing fundamental plasma physics issues like the plasma dynamo problem and the understanding of locked modes.

4.3 Final Statement

As a result of a great deal of progress, a plan for the demonstration of fusion power production in a tokamak has been made. Nonetheless, many scientific issues remain to be perfected and other alternate paths are still likely to develop. The physics issues that need to be resolved and the goal of a long-term energy source remain an inspirational challenge to the technical creativity of mankind.

GLOSSARY

Alfvén wave: A plasma wave that may be excited by alpha particles in a fusion plasma.

burn: The self-sustained heating of a plasma arising from the fusion reaction.

divertor: The edge region of a tokamak where field lines divert escaping plasma away from the immediate vicinity of the main discharge.

energy channelling: A proposed process to allow charged fusion products, with assistance from radio frequency excitation, to heat ions rather than directly heating electrons.

halo current: Currents that are partially carried by the plasma but pass through the surrounding walls.

ignition: The process whereby plasma heating transfers from being externally supplied to being internally supplied from the fusion reactions.

magnetic field ripple: The modulation in the magnitude of the magnetic field due to the discrete number of toroidal field coils.

pedestal: A region of plasma adjacent to the edge where the plasma temperature abruptly rises.

quasi-helical: A method of achieving intrinsic symmetry in the design of three-dimensional stellarator magnetic fields to optimize single particle confinement.

semi-detached: A divertor plasma condition where a large fraction of the escaping plasma has recombined into neutral atoms.

shear flow: Flow of plasma that varies with minor radius.

thermal barrier: A region in a plasma where local transport properties are suppressed.

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Further Reading

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Figure 1: An artist's conception of a tokamak reactor, with cut-out showing plasma discharge.

Figure 2: Cross-section of a typical design of a tokamak for a fusion producing experiment which has been produced by the ITER team.