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COMMISSION OF THE EUROPEAN COMMUNITIES

THE PRESENT STATE OF RESEARCH INTO PLASMA HEATING AND INJECTION METHODS

1974

Report prepared by
the Euratom Advisory Group on
Heating & Injection

LEGAL NOTICE

This report is on sale at the addresses listed on cover page 4

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Commission of the European Communities
D.G. XIII - C.I.D.
29, rue Aldringen
Luxembourg
December 1975

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The heating and injection requirements of four reference reactor designs are previously defined. The problems which arise when one attempts to extrapolate existing work towards the reactor goal are emphasized.

Two refuelling methods not directly linked with the heating problem are discussed. The experiments in operation or under construction in Europe in which each method is investigated are listed.

Sixteen working papers which served as a basis for the Advisory Group discussion and which cover all the heating and injection methods examined are included.

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ABSTRACT

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S E C T I O N A

The Present State of Research into Plasma Heating and Injection Methods

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The membership of the Euratom Heating & Injection Advisory Group (including alternate members) during the two years within which this report was prepared was as follows:

Chairman: C.J.H. Watson - UKAEA Culham Laboratory, Abingdon, Berks., U.K.

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The main work of preparation of the report was carried out in 1973: however the final revisions, and the conclusions reached, are the responsibility of the 1974 membership.

In addition, a large number of experts on the various methods of heating and injection participated in the work. The following experts contributed to working papers submitted to the Advisory Group during 1973.

EURATOM-CEA, Fontenay-aux-Roses	Drs J. Adam, R.A. Dei-Cas, J.P. Girard, F. Bottiglioni, J. Coutant, M. Fois.
EURATOM-CNEN, Frascati	Prof. B. Coppi, Drs A. Cavaliere, F. Engelmann, L. Enriques, F. Santini

EURATOM-FOM, Jutphaas	Drs H. deKluiver, B. Brandt
EURATOM-IPP, Garching	Drs S. Puri, G. Schilling, C. Andelfinger
EURATOM-KFA, Julich,	Dr W. Bieger
EURATOM-UKAEA, Culham	Drs D.R. Sweetman, P. Davenport, D. Robinson, S.M. Hamberger, W. Millar, C.N. Lashmore-Davies, I.J. Spalding, J.G. Cordey, J. Sheffield
Kernforschungszentrum, Karlsruhe	Drs R. Klingelhöfer, W. Henkes

In addition, we benefited from correspondence or discussions at our meetings with: Drs. J.G. Wegrowe, M. Kaufman and C.B. Wharton of EURATOM-IPP, Garching, J.C. Martin of AWRE, Aldermaston, and Drs A. Gibson and R. Hancox of EURATOM-UKAEA, Culham Laboratory.

We would like to acknowledge here our gratitude to all these experts for the (often considerable) time which they put at our disposal, and also to Mr J.L. Hall and the members of the staff of Culham Laboratory who undertook the task of the editing and publishing of the final text.

THE PRESENT STATE OF RESEARCH INTO PLASMA HEATING AND INJECTION METHODS

1. INTRODUCTION

During this decade, the mean energy content of the plasmas in fusion containment experiments will again rise by an order of magnitude. In the largest of the experiments now being designed, the contained energy will be only one order of magnitude short of that required in a fusion reactor: in the Joint European Tokamak, for example, the plasma will have a total energy of 20 MJ. The provision of such large amounts of energy within one containment time implies high power levels, and hence technologically sophisticated equipment. The rising energy requirement will therefore be matched by an increase in expenditure. The rate of investment in research on plasma heating and injection in Europe, which has for some years been at a level of 1 - 2 MUC per annum, will probably have to be increased to meet the needs of JET, and certainly to meet the needs of a fusion reactor programme. There will have to be more coordination of this research, and possibly more concentration on the most promising lines, if the requirements of the next generation of containment devices are to be met.

Against this background, the Euratom Heating and Injection Advisory Group decided that it was opportune to prepare a report on the present state of research in this area, with a special emphasis on the problems which arise when one attempts to extrapolate existing work towards the reactor goal. The present Report is the outcome of that decision. We have attempted to make the discussion as complete as possible: we have considered all the heating and injection methods which are currently under investigation in Europe, we have examined them in relation to four different fusion reactor designs, and we have taken into account not only the energy requirements of a reactor but also its need for some means of refuelling and (in some cases) for some means of maintaining a toroidal current within the plasma.

Our method of work is described in Section B.1: in essence, a small working group of experts was set up for each heating and injection method, and these groups prepared in draft form the various sections of this report. These were discussed at four meetings of the Advisory Group, and revised and approved versions of them appear as Sections B.4 - B.6. In addition, working groups were set up to examine the heating and injection requirements of the four reactor concepts and the constraints which the reactor environment imposes on the design of heating and injection equipment: their reports appear as Sections B.2 - B.3. Finally, for completeness, we asked experts on the reactor refuelling problem to prepare papers on the two refuelling methods which are not directly linked with the heating problem and these appear in Section B.7.

In the light of the evidence submitted to the Advisory Group by these working groups, we have attempted in the following pages to assess the heating and injection requirements of each of the four reference reactor designs, and to sum up the case for and against each method (bearing in mind the rather differing requirements and constraints of the reference reactor designs) and to give the overall judgement of the Group on its prospects. In the first instance, we sought to express this judgement by assigning a letter to each method, using the scheme of classification proposed in Section B.1, and traces of this procedure can still be found in the papers in section B. However in the end we decided that it was more useful to give a very brief summary of the principal advantages and disadvantages of each method, as they now appear to us, and these 'assessments' will be found at the foot of each section of part 3 below and are collected together in the Conclusions on page 25 We also list the experiments now in operation or under construction in Europe in which the method is being investigated.

2. THE HEATING AND INJECTION REQUIREMENTS OF THE FOUR REFERENCE REACTOR DESIGNS

2(a) TOKAMAK

This was taken as an example of a low β toroidal reactor with a plasma initially heated by an ohmic heating pulse and then raised to ignition (and possibly sustained in a steady state) by some supplementary heating or injection method. The supplementary heating requirement is highly controversial, and ranges from zero to 100 MJ depending on the assumptions made. Among the uncertainties are the enhancement of the classical resistivity by neoclassical effects, high Z impurities and anomalous resistivity, the enhancement of the energy loss rate above the neoclassical value (for which there is experimental evidence which is not understood), and the scope for low-density start up, which may be limited by synchroton radiation or by the particle containment time. As regards the enhancement of the resistivity, only a substantial anomalous resistivity would permit ignition at full density, and this brings risks of both enhanced diffusional losses and of a strong skin effect leading to MHD instability. As regards low-density start up,the implied containment time (some tens of seconds) is at the upper limit of credibility. Our assessment is therefore that it would be unwise at this time to rely upon the particular combination of favourable assumptions which lead to a small (< 10 MJ) heating requirement. We have in consequence arbitrarily fixed the requirement at 100 MJ, to be supplied within one second.

2(b) STELLARATOR

This was taken as an example of a low β steady state toroidal reactor, in which the reaction is self-sustaining after ignition. The ignition requirements of this system depend upon the extent to which provision is made for ohmic heating during start-up. If ohmic heating currents at the Tokamak reactor level are assumed to be acceptable, the supplementary heating required is of the same order as that required by the Tokamak reactor, and is subject to the same uncertainties. If for some reason, such large currents are unacceptable, the supplementary heating requirement would be proportionately increased.

2(c) MIRROR

This is an example of a non-self-sustaining reactor, in which power has to be supplied throughout operation at a level which is a significant fraction (at least) of the net output power. The precise requirement depends sensitively on the particle loss rate assumed, and on the means used to reduce or recuperate the energy lost in this way. Neutral injection seems the most plausible method at present, and the requirement may be as large as 3400 A equivalent at 500 keV or as small as 1000 A equivalent at 100 keV depending on assumptions.

2(d) TOROIDAL PINCH

This was taken as an example of a high β fast pulsed reactor. There has been a considerable evolution in the design of such reactors recently and the parameters are still rather uncertain. A typical design requires $\sim$ 1000 MJ delivered in a programmed manner during a pulse of 100 ms overall duration with a 10 second repetition rate. These figures must for the present be regarded as very tentative.

3. THE ADVANTAGES AND DISADVANTAGES OF PROPOSED HEATING AND INJECTION METHODS

3(a) HEATING METHODS WHICH RELY UPON QUASI-STATIONARY PLASMA CURRENTS

3(a)(i) 'TURBULENT HEATING ($\vec{j}$ parallel to $\vec{B}$)

The case for

The required energy can be provided by a credible extrapolation of existing condenser technology.

The transfer of energy to a plasma by this means has been demonstrated: the energy will probably penetrate the whole plasma and the final state of the plasma can be stable.

The case against

The overall energy efficiency is likely to be ~ 30% with little prospect of recovering the losses. The cost of the energy storage system could be high (~ 1000 MUC).

The vacuum wall of the reactor must have insulating gaps capable of withstanding fields of ~ 7 kV/cm.

There may be highly anomalous plasma losses during the brief heating pulse.

There may be a problem in maintaining MHD equilibrium during the pulse.

Assessment

Existing experiments work on rather short time scales and have been successful in producing interesting plasma densities and temperatures: however for economic and technological reasons it is necessary to move towards substantially longer time scales and lower loop voltages, and there remain unsolved problems of losses during the heating.

Existing European Experiments

Turbulently heated Tokamaks at Frascati and Jutphaas; turbulently heated Torsatron at Culham.

3(a)(ii) 'SHOCK' HEATING ($\vec{j}$ perpendicular to $\vec{B}$)

The case for

The required energy can be provided by a credible extrapolation of existing condenser technology.

The method is the only one presently available for filling large high β toroidal devices.

The case against

The overall energy efficiency is likely to be ~ 10%: there is some scope for recovering magnetic field energy. The cost of the energy storage could be high (~ 1000 MUC).

The vacuum wall of the reactor must have insulating gaps capable of withstanding ~ 10 kV/cm.

The method leads to anisotropic non-isothermal distributions: there may be instabilities during the relaxation phase.

Assessment

This method has been successful in producing fusion plasma conditions and has strong advantages as the means of heating the next generation of high-β experiments: however it is only applicable to a few rather specialised reactor concepts and requires either a major break-through in the economic fast storage of energy or for it to play a complementary role in some combined heating system.

Existing European Experiments

Shock heated high β tori at Culham, Garching, Julich; shock heated mirror at Fontenay-aux-Roses.

3(a)(iii) 'ADIABATIC' COMPRESSION

The case for

The required energy can be provided by existing fly-wheel generator technology (or cryogenic energy storage in the future) at very modest cost ($\sim 10^{-2}$ UC/J).

The method has been shown to work in a large number of experiments, some of which have approached fusion conditions (4 keV at 10^{16} cm^{-3}), and it is theoretically capable of closing the ohmic heating gap in Tokamaks, provided that Bremsstrahlung cooling is dominant.

The slow heating rate minimises the risk of streaming instabilities and the fields can enhance kink stability.

No hardware is required inside the vacuum wall.

The case against

The method requires a large (4-6 fold) increase in the magnetic field energy of the containment system over a reactor heated in some other way. The implicit cost per Joule delivered to the plasma is ~ 1 UC/J, although this figure could be reduced to 0.2 UC/J if the plasma can be re-expanded after ignition.

The energy efficiency of the method is only 25% due to leakage inductance and resistive losses.

The method is of little value if the reactor relies upon wall effects for stabilisation - thus high β configurations are effectively excluded.

Assessment

This method is applicable to a few specialised reactor concepts in which it has considerable promise, possibly in combination with other methods: however from an economic standpoint it depends critically on the re-expansion of the plasma column after ignition to fill the containment vessel.

Existing European Experiments

Small Compression Tokamak at Culham.

3(b) HEATING METHODS BASED ON THE ABSORPTION OF ELECTROMAGNETIC ENERGY

3(b)(i) ION TRANSIT TIME MAGNETIC PUMPING

The case for

The method requires RF power at a frequency below 1 MHz: the required power is almost available and certainly credible, with high efficiency.

The losses between RF generator and plasma can in principle be made very small (< 10%).

The modulation of the confining magnetic field need not exceed 0.1%, so theoretically there should be a negligible effect on equilibrium.

The rate of heating is low and uniformly distributed within the plasma and it should not adversely affect the plasma stability.

There is a possibility that a version of TTMP might be used to solve the refuelling problem.

The case against

The only toroidal TTMP experiment to date has encountered a major problem of enhanced plasma loss. Some preliminary theoretical considerations suggest that this problem may become less serious in larger machines.

The method requires a number (2-10) of coils, probably associated with electrostatic screening, situated inside the first wall of the reactor and at an adequate distance (~ 20 cm) from it. There are two major problems: insulation across the terminals (~ 100 kV if there is only one gap) and cooling the coils (possibly requiring heat pipe technology).

Assessment

This method has the advantage over other RF heating methods in that its efficiency increases with plasma density and radius: however heating without pumpout has still to be demonstrated and serious technological problems are raised by the need for coils within the vacuum chamber.

Existing European Experiments

TTMP on stellarator at Culham.
TTMP on Tokamak at Grenoble
TTMP on stellarator at Grenoble/Garching.

3(b)(ii) ELECTRON TRANSIT TIME MAGNETIC PUMPING

The case for

The required frequency $k_{\parallel} v_e$ is in the range 1-10 MHz. The required power is almost available and certainly credible, with high efficiency at (~ 0.2 UC/Watt). As in ion TTMP the losses between RF generator and plasma can in principle be made small (~ 25%).

The interest of the method lies in the fact that at the frequency $\omega \cong k_{\parallel} v_e$ for which strong collisionless damping exists, one of the first magnetosonic resonances ($k_{\perp} a \cong 2.4$), which is a resonance of the bounded plasma-cavity system, gives rise to a strongly enhanced field inside the plasma. The rf power absorption is thereby maximised and, for a given absorption of power by the plasma, the power losses in the wall are reduced with respect to ion TTMP by a factor $\sqrt{v_i/v_e}$.

Heating is uniformly distributed within the plasma and it should not adversely affect the plasma stability (see existing experiments using magnetosonic resonances on Tokamaks, in which however, at present parameters, the heating mechanism is a non-linear one).

The case against

It has not yet been tried experimentally with electron TTMP as damping mechanism. Like ion TTMP, the method requires a number (2 to 10) of coils (or, possibly, of retractable loops) situated inside the first wall of the reactor.

Assessment

This recently proposed method has the advantage that the magneto-acoustic resonance of the plasma toroidal cavity system strongly enhances the wave within the plasma, and the wave is strongly damped when $\omega \simeq k_{\parallel} v_e$; however there are again serious technological problems due to the need for coils within the vacuum chamber.

3(b)(iii) ION CYCLOTRON HEATING (including harmonics of the ion cyclotron frequency)

Note The following discussion refers to heating at twice the ion cyclotron frequency as this is considered to be the most credible heating method for a reactor in this frequency range.

The case for

The method has been used at low power level in a Tokamak and no additional losses were observed.

The acquired RF power (in the 1 metre waveband) is nearly available and certainly credible: the overall efficiency with which it can be produced is ~ 50%.

The power could be fed into the reactor by means of large waveguides, and a partially-plasma-filled torus could act as a resonant cavity in a rather high radial mode number.

According to theory, at these frequencies even linear damping gives an adequate heating rate and the required field strengths are ~ 100 V/cm. (At higher harmonics the waveguide size can be reduced but the cavity resonance condition is less readily fulfilled and it is necessaary to invoke non-linear damping, with a higher threshold field strength.

The case against

The required waveguide has a minimum size of order 1 metre (unless this could be reduced by dielectric loading or shaped waveguide cross section).

The field strength within the waveguide (~ 2 kV/cm) may exceed the breakdown fields under ambient reactor conditions, especially near its junction with the torus.

The energy efficiency of the waveguide coupling may be low, and there may be a heat load problem on the waveguide walls. Dynamic control of the frequency and matching may be required to maintain the optimum efficiency, and would be a severe problem.

The RF generating and frequency tracking equipment is likely to be expensive (~ 1 UC/watt).

The method preferentially heats ions in the transverse direction, and may enhance banana diffusion.

At the required field strengths, parametric instabilities might decrease the containment time.

Assessment

In reactor designs which permit 1 meter access ports, this method has the attractive feature that it is possible to supply the power through unloaded wave guides which could also be used for pumping: however, there may be breakdown problems associated with the high electric fields and there may be a difficulty over dynamic frequency tracking and matching.

Existing European Experiments

Ion cyclotron heating on stellarator at Grenoble/Garching.
Ion cyclotron harmonic heating on linear θ-pinch and toroidal screw pinch at Jülich.

3(b)(iv) LOWER HYBRID RESONANCE HEATING

The case for

The method has been tested at low power levels with no observed increase in diffusion and an efficiency of 50%.

RF power sources at the required level could be developed,on a credible extrapolation of existing technology, at a cost of ~ 0.4 UC/watt. It is possible to use waveguides of moderate (~ 20 cm) dimensions to launch the waves and these could be bent outside the neutron blanket. For credible reactor parameters ($n_e \sim 10^{14}$ cm^{-3}, B_o = 100 kG) the matching problem may not be difficult.

The case against

According to present theory (which is still rather preliminary) the situation on the accessibility and matching problem is as follows:

For axial fields below 100 kG, the maximum accessible density, with no mismatch problems,is restricted to 1 x 10^{14} cm^{-3}, if the grazing incidence approach is used. The energy efficiency of this scheme is ~ 50%, the losses being uniformly distributed over the torus walls.

Higher densities are accessible, but with possible matching problems, using a phased array of waveguides acting as a slow-wave-structure. If a mismatch exists, the losses (~ 40%) will occur within the waveguides.

Both matching and density limit problems can simultaneously be solved by using a passive slow-wave structure but this has difficulties of mechanical construction and cooling and may cause impurity problems.

Assessment

This method fully exploits the advantages of waveguide launching: however its viability depends critically on the accessibility of the resonant surface in plasmas of reactor density. Existing theoretical predictions indicated that this is just possible, but there is only a small margin and there is as yet no experimental evidence.

Existing European Experiments

LHRH on linear machine at Brussels
LHRH on mirror at Garching
LHRH on stellarator at Grenoble/Garching
LHRH on mirror machine at Milan.

3(b)(v) LASER PLASMA HEATING

The case for

The method has already been used to fill a mirror machine with a high β 100 eV plasma and a stellarator with a plasma.

Toroidal reactor power requirements represent a credible extrapolation of existing laser technology: the overall energy efficiency might eventually be 10 - 40%.

No internal hardware is required. Access holes into the reactor could be of $\sim$ 10 cm diameter and could (by contrast with neutral beam heating) incorporate neutron traps.

The method might perhaps be used to accelerate small hydrogen pellets into a steady state reactor for refuelling purposes.

The cost might eventually be low - the figure of $\sim$ 37 MUC per 100 MW delivered to the plasma has been estimated, assuming a 10% heating efficiency. (This corresponds to a cost of $\sim$ 0.025 uc/W(E) of power generated).

The case against

The plasma created initially is far from MHD equilibrium ($\beta \gg 1$) and the dynamics of the resulting expansion are at present a matter for speculation. Polarisation fields or streaming instabilities might lead to rapid plasma loss.

Incomplete ionisation of the pellet, or $\beta > 1$ expansion followed by wall contact, could greatly enhance the background gas pressure.

The present cost calculations assume both a low repetition rate ($\sim$ 100 per second) for the laser and heating near the critical density: it is not clear that these assumptions are compatible unless either laser acceleration of pellets (to ensure penetration) or some combination of low density/small radius start-up is feasible. It is not obvious that the energy requirement is only 100 MJ, unless the method can in some way be combined with ohmic heating. (One possibility is to adapt the 'moving limiter' concept).

Assessment

The attractive feature of this method is the small access requirement and the potentially low cost: however there are major uncertanties about the penetration of the pellets into the plasma and the expansion of the heated pellet.

Existing European Experiments

Laser filling of stellarator at Culham and Garching.

3(c) HEATING METHODS BASED ON THE INJECTION OF ENERGETIC PARTICLES

3(c)(i) NEUTRAL ATOM INJECTION

The case for

Neutral beams with either the current or the voltage required are already available, but not both together, and not yet for very long pulses. The power already available in single units is 200 kW for pulses of 20 ms. The extrapolation to high power units operating quasi-continuously is credible.

The method has already been used to create and maintain a 10 keV plasma at a density of 10^9 in a mirror machine and to increase the ion temperature in Tokamaks by ~ 20% without evident signs of enhanced losses. This increase is approximately what would be expected theoretically.

The energy efficiency could in principle be high if either negative ion sources or efficient direct conversion of the unneutralised portion of the ion beam prove to be practical.

No internal hardware is required: ten holes of ~ 30 cm diameter through the blanket would probably suffice, and would have a negligible effect on neutronics.

Beams of 3 MeV (parallel injection) or 1 MeV (perpendicular injection) should suffice to penetrate a reactor plasma with $n = 3.10^{14}$: these energies are reduced by a factor of 4 if $n = 10^{14}$.

Theoretically parallel injection should not lead to dangerous instabilities, though perpendicular injection might enhance the loss rate due to the large banana orbits or to trapped particle instabilities.

There is a theoretical possibility that parallel neutral injection might be used to maintain a steady state Tokamak equilibrium, especially if the current is enhanced by the theoretical bootstrap effect.

In mirror reactors, neutral injection solves the refuelling as well as the heating problem.

The case against

The energy (3 MeV) required for penetration in the case of parallel injection (which is favoured on stability grounds, and for maintaining toroidal currents) may be technologically difficult (and hence expensive) to achieve at the high current levels required. The possibility of working at lower energies is still speculative. It is suggested that plasma impurities would reduce the penetration depth at a given energy.

The energy efficiency of 1 MeV beams of positive molecular ions without direct convertors is only 20%: the feasibility of high-efficiency direct convertors of acceptable cost has not been demonstrated.

The transport of the neutral beam through the blanket may be impeded by ionising collisions with background gas, which is itself due to re-emission from the walls of the flight tube. An estimate of the resulting current limitation is 20 A through a tube of radius 20 cm.

The straight-line trajectories of the neutrals create a difficulty in screening out fusion neutrons without enclosing the entire hardware (including direct convertors) within a thick biological shield.

The slow gas current associated with the beam represents .03% of the total plasma content appearing at the plasma surface. The consequences are uncertain.

In mirror devices, the plasma is less stable than theoretically predicted during injection: in toroidal devices the injected power has hitherto been a small fraction of the ohmic power.

Even at 1 MeV, the total cost of 100 MW could be ~ 50 MUC: no estimate is available for the extra cost of 3 MeV operation.

Assessment

This method has been operated successfully at the 100 kW level in existing experiments and the extrapolation to reactor power levels is credible: however it is necessary either to develop an acceptable means of penetrating the plasma at ~ 100 keV (e.g. by low density or small radius startup) or to achieve a sufficiently high efficiency when operating at 1 MeV.

Existing European Experiments

Neutral beam development at Fontenay-aux-Roses, Garching, Culham; neutral injection into Levitron, Stellarator and Tokamak at Culham, into Tokamak at Fontenay-aux-Roses, and into Stellarator and Tokamak at Garching.

3(c)(ii) CLUSTER INJECTION

The case for

Generally the same as for neutral injection. However clusters have some specific advantages:

Space charge is less important in the accelerator design because of the 100 fold greater mass/charge ratio.

There is no difficulty in achieving DC operation.

There is no difficulty in neutralising an accelerated cluster with high efficiency.

The associated slow gas flow should be less than from an ion beam neutraliser.

The beam has a naturally spread velocity distribution, which should minimise instabilities.

For a given energy per atom, clusters have a somewhat enhanced ability to penetrate a plasma.

As regards the availability of power at present a 1 MV, 10 keV/atom 10 A equivalent cluster ion acceleration is under construction. The power already available in single unit rectifiers is 500 kW at 2.5 MV steady state.

It is conceivable that injection of relatively large clusters (~ 10^8 atoms/cluster) might solve the refuelling problem.

The case against

In general the same as for neutral injection, except that cluster beams have not yet been injected into a plasma, and so far the power available in cluster beams has been much less than that in neutral atom beams.

There is no immediate prospect of going beyond 100 keV per atom, and even this requires a 10 MV accelerator. Thus the method is only credible, as applied to toroidal reactors, if the penetration problem can somehow be circumvented (e.g. by plasma build-up with a growing plasma radius), and only credible in relation to mirror reactors if 100 keV operation is feasible. The feasibility of 10 MV, 1 Amp equivalent accelerators needs to be established.

Assessment

Unlike neutral injection, this method has not been tested experimentally on a plasma, but it potentially has several advantages (spread velocity distribution, enhanced penetration, possibly higher neutralisation efficiency): however it is limited to 100 keV/atom (10 MV acceleration) and even with two fold enhanced penetration it is tied to a low density or small radius start up.

Existing European Experiments

Cluster source development at Fontenay-aux-Roses, Karlsruhe.
Cluster injection into Tokamak at Fontenay-aux-Roses
Cluster injection into Stellarator at Garching/Karlsruhe.

3(c)(iii) GUN PLASMA INJECTION

The case for

The method has been shown to work in present generation containment devices.

There is a doubtful prospect that the compressed flow device might be developed to meet the reactor energy requirement.

The case against

There is a major problem in transporting the gun plasma into the containment field. Cross field injection leads to polarisation field losses: longitudinal guide fields need to be pulsed in $\sim 25\ \mu s$ and do not solve the problem of access to the centre of the reactor.

The overall energy efficiency is likely to be low.

The cost per joule delivered is likely to be extremely high.

Assessment

This method is not applicable to a reactor unless some unexpected development allows the efficient transport of the injected plasma across the magnetic field.

Existing European Experiments

Hall accelerator at Culham.

3(c)(iv) RELATIVISTIC ELECTRON BEAM HEATING

The case for

The power requirement represents a credible extension of existing technology.

Theoretically and experimentally, turbulence generated by the return current is an effective plasma heating method.

It is conceivable that an injected relativistic beam might be used to maintain a steady state toroidal or multiple mirror configuration (Yoshikawa, Budker).

The case against

Existing technology is already very advanced (military applications) and even the proposed two-fold increase in voltage would be a major step. It is not obvious that further economies of scale are obtainable and existing costs ($\sim$ 10 UC/Joule) are discouragingly high. The proposed value of the anode-cathode spacing (a factor of 2 smaller than Aurora) is controversial.

The energy efficiency of REB production is currently about 50% and heavy losses are normally experienced in transporting the beam over a few metres.

Unless a solution can be found to the beam transport problem, the diodes might have to be placed within the reactor vessel. It is doubtful whether 100 cm holes would suffice to feed the diodes, (the 12 MV coaxial feed of Aurora is $3\frac{1}{2}$ feet in diameter) and the neutronic implications of the hardware might be serious.

The dynamics of the REB within a toroidal plasma are not understood: some elementary arguments suggest that the beam should ignore the magnetic field, but experimentally this does not appear to be the case. For the method to work as proposed, the beam must be confined within the plasma for $\sim$ 100 circuits of the torus.

Assessment

This method is not applicable unless some unexpected development permits both the use of low energy beams ($\approx$ 1 MeV) at high power and the efficient transport of the beam across the magnetic field on the relevant time scale ($\approx$ 10 nsec).

Existing European Experiments

REB plasma heating at Amsterdam.

4. THE REACTOR REFUELLING PROBLEM

Of the four reactor concepts considered here, one (the Toroidal Pinch) avoids the refuelling problem by arranging for the plasma burn-up time to be of the same order as the overall pulse length, and one (the Mirror) solves the problem by operating at a density and mean energy such that the same neutral beam can be used both to heat and refuel it. The other two (Tokamak and Stellarator; and indeed any steady state or quasi-steady state reactor concept in which the temperature of operation is too low for neutral atoms at that energy to penetrate the plasma) have a major refuelling problem. Four outline solutions to be problem are currently being investigated:

(i) Pellet injection
(ii) Large cluster injection
(iii) Gas blanket refuelling
(iv) TTMP pump-in

4(i) The pellet injection approach has the advantage that there is no difficulty in meeting the material flux requirement: its principal disadvantage is that there is an upper limit to the velocity with which such pellets might credibly be injected (10^3 - 10^4 m/s) and at such velocities the pellet would not penetrate to the centre of a reactor plasma unless it were shielded from plasma ablation by a neutral gas layer, plus either electrostatic or magnetic effects. The best available experimental evidence (which is still very preliminary) suggests that the shielding by a neutral gas layer may not be effective. However, the loss rates observed may be affected strongly by the radial electric field present. The experiment does not allow any conclusions to be drawn about electrostatic or magnetic shielding. On existing theory, magnetic shielding should only be effective if the reactor operates at a β-value which is somewhat ($\sim$ twice) higher than the expected value for steady-state toroidal reactors. However, more work is urgently needed to obtain a definitive assessment of this method.

4(ii) The large cluster approach is based on the fact that clusters of $\sim 10^8$ atoms can simultaneously be opaque to 10 keV electrons and transparent to recombination radiation. Such clusters could probably be accelerated electrostatically to 10^4 m/s, and at this velocity might pepetrate a reactor plasma provided that the reduced energy transfer sufficiently decreases the ablation rate. Such clusters can be produced, but so far there has been almost no work on this acceleration or on their penetrating power.

4(iii) The gas blanket approach is based on the argument that even if the probability that a neutral atom will penetrate to the centre of the reactor is low, a sufficiently high surrounding gas density will create density gradients within the plasma such that fuel plasma diffuses inwards and helium outwards.

The principal problem is to establish that there exist parameters for which the transport of energy out of the plasma, which is enhanced by Bremsstrahlung and collisions with the (necessarily high) helium component of the plasma, remains compatible with a favourable reactor energy balance. Existing theoretical studies suggest that such a regime does exist, at least in an infinite cylindrical plasma with classical transport coefficients. The required magnetic field is reasonable; the β value is high, but perhaps achievable. It remains uncertain whether toroidal regimes also exist, in view of the enhancement of the transport coefficients due to toroidal effects. Existing experiments are still rather far from a reactor regime.

4(iv) The TTMP pump-in approach is based on the concept that a net inward diffusion of fuel plasma can be achieved without a very high gas blanket pressure, if suitable TTMP fields are present within the plasma. This method is proposed very tentatively, and even if the diffusion operates in the manner suggested, the problems involved in applying TTMP fields within the reactor environment remain severe.

5. CONCLUSIONS

The present state of research into plasma heating and injection methods does not permit any very definite conclusions to be drawn. Of the twelve methods which are currently under investigation in Europe, the assessment of the Advisory Group is that no method is so well established that it is clearly in principle applicable to the heating of a reactor, and has no foreseeable difficulties; nor can any method be ruled out as completely hopeless. This is not to say that all twelve methods are equally placed, and in this report we have tried to identify both the particular advantages of each method, and the most serious problems which remain to be solved, if it is to be applicable to the heating of a reactor. Naturally, this involves judgement, and in some measure the assessments which we have made are influenced by the particular composition of the Advisory Group; nevertheless we believe that it is useful to present the assessment of a group of 27 physicists, who have examined the best evidence available to them in February 1974. Our assessments are to be found above: for ease of reference we collect them here.

Turbulent Heating. Existing experiments work on rather short time scales and have been successful in producing interesting plasma densities and temperatures: however for economic and technological reasons it is necessary to move towards substantially longer time scales and lower loop voltages, and there remain unsolved problems of losses during the heating.

Shock Heating. This method has been successful in producing fusion plasma conditions and has strong advantages as the means of heating the next generation of high-β experiments: however it is only applicable to a few rather specialised reactor concepts and requires either a major break-through in the economic fast storage of energy or for it to play a complementary role in some combined heating system.

Adiabatic Heating. This method is applicable to a few specialised reactor concepts in which it has considerable promise, possibly in combination with other methods: however from an economic standpoint it depends critically on the re-expansion of the plasma column after ignition to fill the containment vessel.

Ion TTMP. This method has the advantage over other RF heating methods in that its efficiency increases with plasma density and radius: however heating without pumpout has still to be demonstrated and serious technological problems are raised by the need for coils within the vacuum chamber.

Electron TTMP. This recently proposed method has the advantage that the magnetoacoustic resonance of the plasma toroidal cavity system strongly enhances the wave within the plasma, and the wave is strongly damped when $\omega \simeq k_{\parallel} v_e$; however there are again serious technological problems due to the need for coils within the vacuum chamber.

ICRH (multiples of Ω_i). In reactor designs which permit 1 meter access ports, this method has the attractive feature that it is possible to supply the power through unloaded wave guides which could also be used for pumping: however, there may be breakdown problems associated with the high electric fields and there may be a difficulty over dynamic frequency tracking and matching.

LHRH. This method fully exploits the advantages of wave guide launching: however its viability depends critically on the accessibility of the resonant surface in plasmas of reactor density. Existing theoretical predictions indicate that this is just possible, but there is only a small margin and there is as yet no experimental evidence.

Laser Heating. The attractive feature of this method is the small access requirement and the potentially low cost: however there are major uncertainties about the penetration of the pellets into the plasma and the expansion of the heated pellet.

Neutral Injection. This method has been operated successfully at the 100 kW level in existing experiments and the extrapolation to reactor power levels is credible: however it is necessary either to develop an acceptable means of penetrating the plasma at $\approx$ 100 keV (e.g. by low density or small radius startup) or to achieve a sufficiently high efficiency when operating at 1 MeV.

Cluster Heating. Unlike neutral injection, this method has not been tested experimentally on a plasma, but it potentially has several advantages (spread velocity distribution, enhanced penetration, possibly higher neutralisation efficiency): however it is limited to 100 keV/atom (10 MV acceleration) and even with two-fold enhanced penetration it is tied to a low density or small radius start up.

Plasma Gun Injection. This method is not applicable to a reactor unless some unexpected development allows the efficient transport of the injected plasma across the magnetic field.

Relativistic Electron Beams. This method is not applicable unless some unexpected development permits both the use of low energy beams ($\approx$ 1 MeV) at high power and the efficient transport of the beam across the magnetic field on the relevant time scale ($\approx$ 10 nsec).

Reactor Refuelling. As regards the reactor refuelling problem, the situation is very unclear, and we do not believe that sufficient effort is being devoted to its clarification by the fusion community. Of the four methods considered here, only two are currently being investigated experimentally within Europe, and in neither case will the existing experiments provide the basis for a firm decision on the applicability of the method. Because of the possible link between heating and refuelling methods, we do not believe that it is premature to expand this work, and our discussions have indicated that there is a lot of scope for European collaboration on this problem.

S E C T I O N   B

Working Papers Submitted to the Advisory Group

TERMS OF REFERENCE GIVEN TO THE WORKING GROUPS

At the meeting of the Heating and Injection Advisory Group held at Brussels on 5-6 March 1973 it was agreed that the time was ripe for the production of a European Status Report on the physical aspects of plasma heating and injection methods on the grounds that:

(i) It was a necessary preliminary to the formulation of an overall European research programme on Heating and Injection that the members of the Advisory Group should have a coherent view of the field, especially of the questions which remained unanswered and the gaps in the existing programme.

(ii) There was an urgent need for the assembly of the best available information on the methods which were likely to be applicable to the next generation of containment devices (for example the Joint European Tokamak), because of the constraints which heating and injection methods might impose upon their design.

(iii) There was a growing pressure within the Controlled Fusion programme to show that methods used on existing experimental devices could be extrapolated to the conditions which would obtain in a fusion reactor.

It was therefore decided that a Status Report should be compiled which would reflect the present state of informed European opinion on all the heating and injection methods which are now under investigation. For each method, one expert was assigned the task of coordinating by correspondence the work of a small working group, which would produce an agreed draft of one section of the Status Report. These draft reports were then circulated to all members of the Advisory Group and to a number of other experts. The present report consists of these sections, as modified in the light of discussions held at subsequent meetings of the Advisory Group.

In order that due emphasis should be put on the large experiment/reactor orientation of the report, each working group was asked to bear in mind the following criteria and questionnaire.

Criteria and Questionnaire for the Comparison of Heating and Injection Methods (drawn up by C.J.H. Watson)

In order to make a fair comparison between the various proposed methods for plasma heating and injection in large experiments and fusion reactors, it is important to have a standard set of ground rules to apply and an agreed framework for expressing the results of the comparison. Since the latter is inevitably somewhat subjective, the following set of categories seems to be as precise as is useful at this stage. The method might be judged to be:

A) Clearly in principle applicable: no foreseeable difficulties

B) Applicable provided that a few physical or technical difficulties can be overcome

C) Difficult to assess in the light of present knowledge

D) Inapplicable unless some unexpected breakthrough occurs

E) Clearly hopeless

To place each of the various methods in one of these categories, the following general questionnaire seems relevant. (For ease of use, repeated as pull-out supplement at end.)

1) Is the required heating power available in this form, or at least credible?
2) How efficiently can power be created in this form?
3) How small can one make the losses in transit between source and plasma?
4) Are the consequences of the unavoidable losses acceptable (e.g. wall heating, degassing, sputtering)?
5) Is there an associated gas load problem?
6) What holes are required in the vacuum wall and are they compatible with neutronic requirements.
7) What hardware is required within the vacuum chamber; will it function in a reactor environment and what are its neutronic implications?
8) How does one ensure that the power permeates the whole plasma?
9) What fraction of the incident power is absorbed by the plasma and what happens to the remainder?
10) Do any electric or magnetic fields associated with the method affect particle containment directly?
11) What plasma distribution function results: is it stable, and if not do the instabilities affect plasma containment?
12) Can one estimate in order of magnitude the capital cost of all the hardware required per kW delivered to the plasma?
13) To which reactor concepts is the heating method applicable?
14) If none, is there nevertheless a case for using the method in the next generation of experiments?
15) Does the method also solve the reactor refuelling problem?
16) Does the method also maintain a plasma current which significantly improves the containment properties of the magnetic field?

HEATING REQUIREMENTS OF VARIOUS REACTOR CONCEPTS

by

J. Adam, D. Sweetman, C.J.H. Watson

1. CLOSED CONFIGURATIONS

A closed configuration reactor of the Tokamak-Stellarator type will be operated at a mean plasma density of $\sim 3.10^{14}$ cm^{-3} and a mean temperature around 20 keV, corresponding to a total energy content of about 500 MJ. However, thanks to efficient heating by α particles above a temperature of a few keV, the problem of supplying power from outside is simply to raise the plasma to ignition temperature during the initial phase of operation.

Various authors have considered the consequences of the power balance equation in this case, in which the power supply is due to ohmic heating, α particles and a supplementary heating, the losses resulting from thermal conduction, Bremsstrahlung and synchrotron radiation.(1)(2)(3)(4)

Although slight differences appear in the results, due to different assumptions in the codes used, the conclusions can be summarised as follows:

- On the basis of optimistic assumptions on the behaviour of the plasma, (thermal losses $\leqslant$ 4 times neoclassical, $Z_{eff} = 1$) and provided the plasma dimensions are large enough ($R \sim 1{,}000$ cm) ohmic heating alone seems marginally capable of bringing about ignition if the plasma density is lower than $\sim 3.10^{13}$ cm^{-3}, which is an order of magnitude lower than what is required for economic operation of the DT reactor. However in this case, the additional heating power required to achieve ignition at a density of 3.10^{14} cm^{-3} is only 100 MW (~ 0.1 W/cm^3), a rather modest value as compared to the 5000 MWe produced by such a reactor.
- If $Z_{eff} = 2$, a supplementary heating is required at any density of operation, due to the increased Bremsstrahlung radiation. A small amount of additional power, however, brings about impressive changes in the evolution of plasma parameters at low density. As shown in Fig.1 (from Stix), ignition can be attained at 3.10^{13} cm^{-3} if an external power supply of ~ 10 MW is available.
- Similar conclusions are valid in the case of an ignition demonstration experiment(4). In that case, ignition could be achieved at low density at the expense of an amazingly small amount of additional power ($R = 200$ cm, $q = 3$, $B = 73$ kG, $\chi_i = 4$ times neoclassical, $P_{add} < 1$ MW).

If those conclusions are accepted, the problem of reaching the conditions appropriate to ignition at low density appears as a relatively easy one. However, four remarks have to be added:

- The temperature achieved at low density is mainly limited by synchrotron radiation and should thus be viewed with caution, as pointed out by Sweetman[4], because of the limitation in the present synchrotron calculations.
- The energy containment times implied in those calculations are inversely proportional to the density and become rather long at low density (several tens of seconds). The contribution of any anomalous losses (impurities, turbulence) would require working at larger density, the necessary supplementary heating power being correspondingly larger. Moreover, if the reactor is pulsed, the duty cycle of course sets an upper limit to the time during which the additional heating power needs to be supplied.
- Assuming low density ignition has been achieved, an appropriate way of raising the density would be required, in order to reach the conditions for reactor operation. One way of overcoming that problem has been suggested by Girard et al[5] using neutral injection in a plasma limited by a limiter of expanding radius.
- Stellarators are only equivalent to Tokamaks if it is assumed that in both cases the same level of ohmic heating is employed. If for some reason (e.g. stability) it were necessary to limit the ohmic heating current in a stellarator, the supplementary heating requirement would be increased correspondingly.

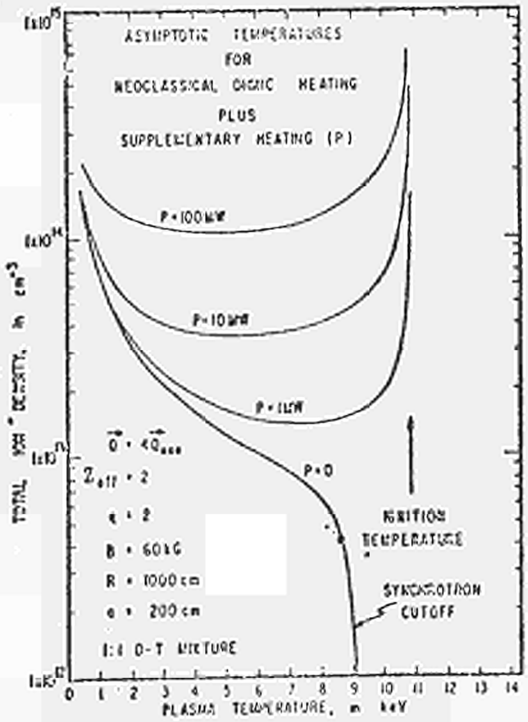

Fig.1 Effect of supplementary heating on reactor size Tokamak for Z_{eff} = 2 (Stix[3]).

2. MIRROR CONFIGURATIONS

A reactor based on the mirror concept has to be operated at a plasma temperature an order of magnitude higher than in a closed configuration.

Since heating by α particles is inefficient, the reactor cannot be self-sustained and a continuous flow of power has to be provided from outside to compensate the particles and energy losses. Neutral injection appears as the most appropriate way of supplying the density and energy.

Computation by Cordey et al[6] of optimized mirror reactors based on the principle of energy recovery suggested by Post[7], assuming that the direct convertor has a fixed cost per kW handled, shows that the optimum injection energy is around 100 keV. However, more recent work[8] has shown that the cost of the direct convertor is a rather steep function of the injection energy, so that the true optimum energy for such a system is likely to be closer to 500 keV. The power handled by the injector in such systems is of the same order as the power output of the reactor: 8700 A at 100 keV or 3400 A at 500 keV for a DT 1000 MWe unit. This study assumed that the Q value of the mirror was given by $Q = 2 \log_{10} R$, where R is the mirror ratio. This may be an underestimate[9], and the injector power requirements fall rapidly with increasing Q.

A number of schemes have been proposed to reduce the end losses which lead to this large injector requirement. One such scheme is the "auto-injection" mirror suggested by Cordey et al[6]. Others are the "toroidally-linked" mirror[10] and the wetwood burner[11]. All these schemes lead to a significant enhancement of Q and a corresponding reduction in the injector current (1000 A) and a relaxation in the required energy (down to 100 keV).

3. HIGH β FAST PULSED REACTORS

As compared to steady state or long pulsed reactors, a fast pulsed system differs considerably: the instantaneous heating power provided by adiabatic compression is several orders of magnitude larger than the power obtainable from other methods (ion injection, RF) and brings the plasma at a density of 10^{16} cm^{-3} to thermonuclear temperature in ~ 10 ms.

Burnett et al[12] have calculated the requirements for a 1.75 GWe θ-pinch reactor operated during 100 ms once every 10 s: the total energy spent in "preparing" the plasma is the initial plasma thermal energy (2.26 MJ/m), the losses in the compression coil and blanket (6.06 MJ/m) and the losses in the cryogenic magnetic energy storage system (1.40 MJ/m). A large part of this is recovered by direct conversion during the plasma expansion (6.14 MJ/m) so that the total energy required per cycle amounts to 3.58 MJ/m or 1250 MJ every 10 s (7% of the output power).

4. CONCLUSIONS

(a) If the full density of 3.10^{14} cm^{-3} is required to ignite a Tokamak-Stellarator plasma, the heating power required in addition to the Joule heating is ~ 100 MW if $Z_{eff} \sim 1$ for a 5 GWe reactor. This power could however be substantially lowered if the problems associated with low density ignition can be solved.

(b) In the case of mirror devices, most of the power generated has to be recirculated through the reactor. At first sight, it seems that the work should be handled by powerful neutral injectors. Several schemes exist, however, which would lower the power injected from outside down to $\sim 10\%$ of the power output.

(c) High β reactors require high energies during short pulses. In a typical reactor producing 5 MWe/m, the energy pulse required is 3.6 MJ/m every 10 seconds.

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CONSTRAINTS UPON PLASMA HEATING AND REFUELLING CONCEPTS IMPOSED BY THE REACTOR ENVIRONMENT

by P A Davenport

1. INTRODUCTION

The most important constraints arise from the fact, often underemphasised, that the reactor environment comprises several nested zones in which the ambient conditions become increasingly hostile and unfamiliar as one moves towards the reacting plasma. In summary:

ZONE	TEMPERATURE ^{o}K	NEUTRON FLUX n/cm^2 sec
Enclosure (inside bioshield)	290	10^8
Magnet	4	10^8
Magnet shield	350	$2 \times 10^{13} \rightarrow 10^8$
Blanket	850	$2 \times 10^{15} \rightarrow 2 \times 10^{13}$
Vacuum chamber	10^9	3.6×10^{15}

All apparatus for plasma heating and refuelling must be capable of long term reliable operation in the ambient conditions of the zones it spans, and this without human intervention there.

In addition to these general constraints, which are applicable to most reactor concepts, these are a number of more specific constraints arising from environmental aspects which can be closely quantified on economic and engineering considerations. As an example, the minimum linear dimensions of a reactor are largely dictated by economics. This has repercussions on the feasible values of such parameters as neutral injection energies and rf heating wavelengths. Furthermore, the choice of the first wall material and its constructional form is likely to be dominated by mechanical, thermal and neutronic considerations. This imposes constraints on rf heating schemes because the effective resistivity of the wall may result in unwelcome power losses. Thirdly we may cite the required thickness of the blanket and shield which is determined by considerations of tritium breeding and mechanical integrity. All power transmitted to the plasma must penetrate this thickness, which places weak upper limits on hole dimensions and power transfer efficiencies.

These and the effects of other constraints are considered in more detail below; appended is a table giving some parameters important to considerations of plasma heating and refuelling for typical examples of four reactor concepts:

(a) Stellarator (steady state)
(b) Tokamak (slow pulse)
(c) Mirror (steady state)
(d) Closed high-β (fast pulse)

2. CONSTRAINTS DUE TO LINEAR DIMENSIONS

2.1 Neutral Injection

The plasma radius is one factor governing the optimum energy for neutral injection, for both heating and refuelling the plasma. Riviere[1] has calculated the depth of penetration of fast hydrogen atoms into a fusion reactor plasma. He concludes that for a toroidal reactor with $n = 3 \times 10^{14}$ cm^{-3}, T_e = 20 keV and a plasma radius of 125 cm, a deuteron energy of 1 MeV is required. This constraint, and the possibility of easing it by operating at a lower starting density, are examined in detail in sections B.2 and B.6(a).

These figures represent upper limits; should it prove possible to exploit anomalous transport processes, resulting for example from turbulence, then they could be significantly reduced.

2.2 Coupling rf power to the plasma

It has been suggested that the vacuum vessel and plasma could themselves be matched to an rf generator, thus avoiding the use of internal coils. (See section B.5(b)).The problem is that of the efficient coupling of the generator to the load constituted by the plasma surrounded by the outer wall. In principle any load may be matched to the generator using an external network matching system which (i) neutralises the reactance presented by the load and (ii) adjusts the coupling to the load in order for the feeder to be matched at its characteristic impedance; but if the mismatch is large, the losses in the tuning network are high due to large standing waves in the tuning system. Dr A Messiaen points out that tuning and coupling would be greatly eased if one were to use a frequency near an eigenmode of the system constituted by plasma, vacuum and outer wall. In practice, such resonances exist:

(a) In the domain of the Alfvén waves $|k_{/\!/} V_A| \leq \omega \lesssim \omega_{LH}$ where they may coincide with ω_{ci} or $n\ \omega_{ci}$

(b) When coaxial modes are excited for $k_o \geq |\ k_{/\!/}\ |$; then the plasma plays the role of the central conductor of a coaxial cavity.

3. RF LOSSES IN THE FIRST WALL

3.1 Material

The choice of material for the first wall is likely to be dominated by mechanical, thermal and neutronic considerations. Its electrical resistance at operating temperature may give rise to significant rf losses. For example, niobium - 1% zirconium alloy, a preferred wall material, has a conductivity of 2.7×10^6 mho/metre at 500^oC (ie about twenty times less than that of copper at 20^oC), and a skin depth δ of 3.1×10^{-3} cm at $\lambda = 3$ metres. The maximum Q value of a vacuum chamber of this material considered as a toroidal resonator is approximately $r_{wall}/2\delta$, ie of the order of 10^4. This figure could be much reduced by access holes in the wall.

3.2 Cellular construction(2)

The rf losses in such a resonator would be further increased if, as seems likely, the wall were constructed of cellular modules. Current path lengths in the wall would be increased by a factor of about five. These losses could be reduced if the interstices between the cellular modules were effectively $\lambda/2$ deep, a suggestion due to D J H Wort. The importance of these losses is that they increase the rf power requirements; they do not constitute a heat transport problem providing that a heating efficiency of at least 10% is achieved.

4. CONSTRAINTS DUE TO THE BLANKET AND SHIELD DESIGN

4.1 Access to the plasma

The interplay of tritium breeding and engineering requirements gives rise to two restrictions in a realistic reactor design, a lower limit on the thickness of the blanket and shield and an upper limit on the area available for access to the vacuum vessel for pipe work, injection and pumping ports etc, termed the access fraction(3). Typical values of these quantities are 2 metres and 0.1 respectively, and since many access holes are required, their maximum allowable diameter will here be assumed to be 50 cm. This is a weak upper limit; more stringent constraints arise from the presence of lumped field coils (see Para 5 below). Some consequences of the need to transmit all power to the plasma through the blanket and shield without interference with its integrity or breeding follow.

4.2 Transmission of rf power through the blanket

Power at metre wavelengths will need coaxial transmission lines, since waves in 50 cm tubes would be evanescent.

Power at centimetre wavelengths can be fed through circular wave guides. For λ = 10 cm and a maximum electric field of 10^4V/cm, a 7.5 cm copper tube will handle 1.6 MW with a loss 10 kW/metre length, or 5 W/cm^2, providing close matching is achieved. Thus the power loss in the guide will cause no heat removal problems. The electric field of 10^4 V/cm is a working value for dry air at atmospheric pressure; the breakdown strength in the partial vacuum at the exit of the guide may be much lower and could give rise to a serious design problem at megawatt power levels.

4.3 Internal rf structures

The presence of rf structures (coils, screens etc) within the vacuum wall can only be tolerated provided:

(a) they are cooled to preserve their mechanical integrity,

(b) their contamination of the plasma by sputtering is acceptable,

(c) their effect on the neutron economy is compensated for,

(d) they are located outside the diverted flux surface and so do not act as limiters,

(e) they do not necessitate such an increase in first wall radius as would lead to either an economically unacceptable increase in blanket and magnet dimensions or a technically intolerable increase in plasma-wall distance.

All these factors are calculable; (c) is effectively an additional drain on the access fraction and (e) might be mitigated by having local bulges in the wall.

In addition, electrical insulation is required which is capable of withstanding the environment at the first wall. The electrical conductivity of flame sprayed Al_2O_3 has been measured as a function of neutron dose and temperature in a TRIGA reactor(4), and linear extrapolation of these results suggest that it would be a reasonably good insulator in a fusion reactor environment.

If the sole function of the internal rf structures is to ignite a reactor which will then operate continuously, there would be many advantages to be gained by retracting the structures after ignition, as an aircraft undercarriage is retracted after take-off. This stratagem would reduce the plasma contamination and neutron absorption due to the structures and might even obviate the need to cool them.

5. CONSTRAINTS DUE TO THE FIELD COILS

5.1 Access holes

The presence of discrete field coils places geometric constraints on the location and orientation of access holes. Existing Tokamak reactor designs (at Culham) just allow neutral injection at an angle of 30° to the plasma axis with a maximum magnetic ripple of 2%, the allowable beam divergence being 5°.

The limitation on access hole diameter to about 50 cm gives rise to pumping problems in removing the background gas from the holes to prevent neutral beam attenuation there; this is discussed in Section B.6(a).

5.2 Stray fields

The field coils produce intense local magnetic fields prejudicial to the operation of ion sources and rf generators. Counter measures will cause an additional drain on space and an increase in complexity.

5.3 Containment field

The main containment field permeates the blanket and the possibility that it could cause Lorentz ionisation of neutral beams passing through access holes has been examined. Preliminary studies[5] indicate that this effect is not serious, less than 1% of the beam being lost.

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Appendix: Parameters of Fusion Reactor Designs

(For details of references A1 - A4 see previous page)

PARAMETER	UNIT	STELLARATOR (steady state) Ref. A1.	TOKAMAK (slow pulse) Ref. A2.	MIRROR (steady state) Ref. A3.	HIGH-BETA (fast pulse) Ref. A4.
POWER OUTPUT					
Thermal	GW			2.23	3.6
Electrical	GW	3.2	2.5	1.0	1.8
DIMENSIONS					
Major radius	m	7	12.5		
Circumference	m		78.5		350
or length	m			21.8	
Wall radius	m		2.5	3.86	0.3
Wall area	m^2		1234		
Plasma radius	m	1.75		2.73	.09 - .17
Plasma volume	m^3	420			
WALL LOADING					
Total	MW/m^2	13	4.6	15	3.5 MW/m^2
Neutron flux	n/cm^2s		1.4×10^{14}		88.4 MJ/m
Bremsstrahlung			10 W/cm^2		1.08 MJ/m
Synchrotron					
γ back shine			50 W/cm^3		
Wall material			Nb - 1% Zr		ceramic/Nb composite
Conductivity	$(\Omega\ m)^{-1}$		2.7×10^6 (500°C)		
PLASMA HEATING					
Power or Energy			30 MW	1224 MW	5.93 MJ/m
Duration			up to 1 min.		
DUTY CYCLE					
Duration	sec		> 10 x containment time		0.10
Frequency	Hz				0.10
BREEDING GAIN					
Lithium			0.3		0.07
Flibe			0.09		
Access fraction			15% (Li) ~ 5% (Flibe)		

OHMIC AND ANOMALOUSLY ENHANCED OHMIC HEATING

by

M. Haegi, D. Robinson

(Consultants: B. Coppi, F. Engelmann, L. Enriques, F. Santini)

INTRODUCTION

Is ohmic heating by itself sufficient to achieve ignition in a fusion reactor like a Tokamak or a Stellarator?

In order to answer this question, estimates and numerical calculations have been performed with the following assumptions:

(a) the ohmic heating power has been computed on the basis of a resistivity which can be:

(1) classical

(2) neo-classical

(3) enhanced by high Z impurities

(4) anomalous.

(b) The dominant losses that have been considered are:

(5) Bremsstrahlung

(6) heat conduction due to the ions

(7) heat conduction due to the electrons

(8) cyclotron radiation

(9) high Z losses.

NEOCLASSICAL RESISTIVITY

The existence of trapped and passing particle orbits in the above devices, enhances [9][10] the classical Spitzer resistivity:

$$\eta \text{ neoclassical} = \gamma \eta \text{ classical}$$

$$\gamma(r) = [1 - 1{,}95\ (^{r}/_{R})^{\frac{1}{2}} + 0{,}95\ (^{r}/_{R})]^{-1} \text{ for } Z = 1$$

where r and R are the small and large radii of the torus.

If the resistivity is averaged over the plasma cross section, the neoclassical resistance of the plasma ring would be about 2 times (or even more [2]) above the classical one and consequently would increase the ohmic power given to the plasma by about the same factor.

HIGH Z ENHANCED RESISTIVITY

The presence of high Z ions in the plasma originating from the limiter or from the wall would increase the effective charge Z_{eff} considered in the classical resistivity, which could be defined:

$$Z_{eff} \equiv \frac{\Sigma(Z^2 n_i)}{\Sigma(Z n_i)} .$$

In present day devices, Z_{eff} is not less than 2-4. On the one hand, the surface to volume ratio decreases by one order of magnitude going to reactor size, and consequently the Z_{eff} may tend to decrease. On the other hand, the thermal loading of the wall and the current duration will increase by several orders of magnitude, which will tend to increase Z_{eff}. From the theoretical data available at this time, we know that, to avoid the ignition temperature exceeding 10 keV, the Z_{eff} should not (1) exceed about 2.

The ohmic power would in this case be about 3 times above the classical Z = 1 case, for the same electronic temperature.

ANOMALOUS RESISTIVITY

Once the plasma current is nearly stationary the electron drift velocity V_{De} will be substantially lower than the sound velocity V_s in the case of a reactor size plasma:

$$V_{De} \ll V_s .$$

Theory (11) and experiment (12) show that current induced anomalous resistivity in this condition is negligible.

Plasma turbulence connected for example, with electron trapping in large amplitude waves (13)(5) could lead to anomalous resistivity, but at present there are no theoretical nor decisive experimental results (7) available. In fact the resistivity measured up to now in Tokamaks can possibly be satisfactorily explained (6)(7) by means of the neoclassical model, taking into account the measured Z_{eff}. It is possible to think that anomalous resistivity could be produced by exciting turbulence or plasma waves into the plasma. At the present time it is not clear that the power necessary to induce the turbulence could be of the same order of magnitude as the ohmic power. In addition anomalous plasma losses might be associated with the induced turbulence.

We conclude that anomalous resistivity heating for $V_{De} \ll V_s$ is difficult to assess in the light of the present knowledge.

During the initial phase of plasma and current build-up, the plasma is still cold and:

$$V_{De} > V_s .$$

Anomalous resistivity from the turbulence arising from one or the other of the many current-driven instabilities can take place. This situation has been theoretically and experimentally much more investigated (3)(12)(14) than the previous one. In these conditions the resistivity can increase by a factor $\gg 1$, and could explain the rapid diffusion of the current from the edge to the centre of the plasma cross section in present day devices, where the neoclassical theory predicts a strong skin effect. In the case of a reactor, on the basis of some calculations (15), even this initial phase anomalous inward current diffusion seems not to be sufficient to avoid intolerable skin effect.

Other calculations (16) suggest that this is not necessarily the case. For these reasons a moving limiter has been proposed.

The loss mechanisms that have been used in the different models are relatively well known, but some of them still leave a considerable degree of uncertainty:

- the Bremsstrahlung (5) is proportional to $Z^3 n_i$, but the value of Z is not known, cf. reference (3).
- The heat conduction due to the ions (6) is found to be, for present day machines, 2-4 times the predicted neoclassical value.
- Similarly the heat conduction due to the electrons is $\sim$ 30 times the neoclassical value and possibly several times the pseudoclassical value.

Furthermore, it should be kept in mind, as pointed out by S. Yoshikawa(8), that according to the results in levitrons, in a much lower temperature range, the particle confinement time follows roughly the neoclassical model only below some critical value of the temperature, but decreases for higher temperatures, parallel to the Bohm curve, which might indicate a fundamental difficulty for the confinement of hot plasmas. This would mean that the confinement time of the plasma may be limited to 200-300 times the Bohm time.

CONCLUSIONS

In the optimum case of a pure neoclassical plasma, ignition, with pure ohmic heating seems marginally possible for a density of $\sim 3.10^{13}$ cm^{-3} which is several times lower than the operating density (17),(1),(4),(18),(19),(20).

The heating time would then be about an order of magnitude longer than at the operating density. Additional plasma would then have to be heated by means of the produced α-particles.

An additional heating power of the same order as the neoclassical ohmic heating power of about 5 MW, would relax the previous assumptions to more realistic values; ($Z_{eff} = 1 \rightarrow 2$, $D = 1 \rightarrow 4D_{neoclassical}$; enhanced losses above a critical temperature, etc.) in order to obtain ignition at the above density level.

An additional heating power, an order of magnitude higher, would be necessary to reach ignition in these more relaxed conditions at full operating density. Furthermore, additional heating power could substantially reduce the heating time.

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TURBULENT HEATING ($j \parallel B$)

by

S.M. Hamberger, H. de Kluiver, and H.W. Piekaar

1. INTRODUCTION

It has been demonstrated in a number of quite diverse experiments that very effective plasma heating results from the controlled excitation of turbulence resulting from the relative ion-electron drift in currents either along or across the confining magnetic field. This is concerned with the former case, for which most information is available from experiments in both open and closed systems.

2. HEATING BY LARGE CURRENTS PARALLEL TO B[(1)-(6)]

The necessary large electric field E is applied to the plasma

(a) between electrodes in linear devices

(b) by induction in toroidal devices.

For the current not to be limited by binary collisions it is necessary that $E > E_c$, the critical field for runaway. For suitable turbulence to develop the current density must be such that the drift velocity $v_d > C_s$: this leads to a spectrum of ion-waves (e.g. ion-acoustic) which causes both species to become heated. It is possible that other forms of turbulence (e.g. electromagnetic) occur which are responsible for the details of the heating processes, but the low plasma conductivity itself seems to result purely from electrostatic fluctuations. Electron temperatures > 30 keV (at $n \sim 10^{18}$ m^{-3}) and ion temperatures up to 2 keV (at $n \sim 10^{20}$ m^{-3}) have been reported, the details depending on the duration of the heating pulse (the ions seem to be heated somewhat later than the electrons). Energy densities $\sim 10^{23}$ eV m^{-3} are achieved in typical experiments, and are usually limited by the equilibrium properties of the trap. There is some evidence that the maximum ion temperature increases with B, although the conductivity itself is insensitive to B.

An extrapolation of the physical possibilities to use this method for ignition of JET is described in reference 2. This reference is to serve as §3 of the present paper.

3. "The heating of a tokamak plasma to ignition temperatures by current-driven turbulence", by H. de Kluiver, Rijnhuizen Report 73-76.

4. APPLICATION OF TURBULENT HEATING TO A TOROIDAL REACTOR

Following the same calculating procedure as described in §3 for JET, a list of parameters pertaining to a toroidal reactor based on the dimensions given in reference (7) by Gibson et al. is presented in Table 1.

Two values of anomalous conductivity have been taken to illustrate the dependence of the final parameters for instance, the total voltage around the torus, etc.

5. REQUIREMENTS

It is obvious that regardless of the heating method used, to heat a 400 m^3 plasma to temperatures of $T_e = T_i = 10$ keV, requires a total energy of nearly 400 MJ. The efficiency of the turbulent heating method, defined by $nk(T_e + T_i)V_p/\frac{1}{2}CV^2$, will be $\sim 30\%$ implying a total installed capacitor bank energy of 1.4 GJ. The necessary high loop voltage V_C can be achieved by subdividing the circumference of the torus into an appropriate number of segments so that the capacitor bank voltage can be reduced to a convenient value ($\lesssim 100$ kV). The capacitors which constitute the bank are then of the standard cost* type.

All relevant toroidal experiments use insulating liners, thus avoiding the problem of break-down at the insulating gaps. If, as seems likely for reasons of cleanliness, the first wall must be metal, either multiple insulating gaps or some form of very thin conducting liner will be necessary, which may require development.

6. QUESTIONNAIRE (See pull-out supplement at end; extract from Section B.1)

1. Yes. A capacitor bank of required dimensions can be constructed. In order to get the necessary field strength a toroidal experiment has to be subdivided into a convenient number of sections (see reference 2).
2. The electrical efficiency for energy storage of capacitor bank is almost 100%.
3. See reference 2, 30%.
4. The effects of extra losses in the plasma as runaway currents of untrapped electrons or perhaps, anomalously enhanced diffusion during turbulent heating have to be investigated in prior experiments, but are expected to be small.
5. No gas load problem.
6. No holes in an insulating vacuum wall. In a metallic vacuum wall gaps of 2 cm width around the minor circumference have to be made.

---

* The price of the power installation will be considerably reduced if inductive energy storage as described by Wipf(8) would become available, provided this method can be used for fairly fast ($\sim 10^4$ Hz) discharges.

7. None.
8. Current diffusion proceeds rapidly due to anomalous resistivity; heat seems to diffuse with the ion-acoustic velocity to the centre[6].
9. It is believed that essentially all power transmitted to the plasma can be thermalized.
10. Anomalous diffusion will probably be the worst loss during the relatively short heating time. However, the skin current may be hydromagnetically unstable to short wavelength modes.
11. A stable distribution results in velocity space. It is not known whether the transient phenomenon of the skinlike current profile is stable or not.
12. 0.2 - 0.5 dollar/Joule.
13. To linear and toroidal reactors.
14. --.
15. The answer is no.
16. Since the drift velocity has to exceed c_s for any strong turbulence to take place, after terminating turbulence the current decays downward to $n_e e c_s$. In this method the plasma current used for heating is essential but will have to decay to a lower value to increase q from say 1 to 3.

σ	$T_i = T_e$	n_p	R	a	V_p	$\varepsilon = \frac{a}{R}$	β_{pol}	q	B_T	B_P	I_c	$V_p \times 3\, n_p kT$
10^3	10 keV	2×10^{20} m^{-3}	7.5 m	1.6 m	379 m^3	0.21	1	2.4	14.3 Tesla	1.27 Tesla	3×10^7 A	364×10^6 Joule
2×10^3	"	"	"	"	"	"	"	"	"	"	"	"

	$c_s = \sqrt{e \frac{T_e}{m_i}}$	$u = 1.08\, c_s$	A_p	R_{pl}	L_{sh}	$L_{pl} - L_{sh}$	heating time: $\frac{1}{2} f^{-1}$
	6.5×10^5 m/s	7.0×10^5 m/s	1.35 m^2	3.5×10^{-2} Ω	15.3×10^{-5}. H	2.4×10^{-5} H	23 μs
	"	"	"	1.7×10^{-2} Ω	/ "	"	46 μs

	frequency: f	C	V_C	$\frac{1}{2}\, C\, V_C^2$	$nkT / \frac{1}{2}\, C\, V_C^2$	$\delta = \frac{1}{\sqrt{\mu_o \sigma f \pi}}$	A_δ
	1.1×10^4 Hz	66 μF	7.4×10^6 V	1.80 GigaJoule	20%	0.15 m	1.43 m^2
	5.5×10^3 Hz	264 μF	3.7×10^6 V	"	"	"	"

TABLE 1

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SHOCK HEATING (FAST MAGNETIC COMPRESSION HEATING: $j \perp B$)

by
K.H. Dippel and M. Keilhacker

1. Introduction

Recent research on heating of low density plasmas in high-voltage theta pinches (U_o = 100-500 kV, electric fields are of the order 1 kV/cm) has demonstrated that ion temperatures in the 10 keV range at densities between 10^{13} and $10^{14}\,cm^{-3}$ can be obtained by fast magnetic compression of a deuterium plasma in times of the order 1 μs[(1)-(6)]. The heating is mainly by shock waves with only little adiabatic compression resulting in a weakly compressed plasma column.

2. Implosion dynamics and compressed plasma parameters

As is well known the magnetic piston that accelerates and compresses a low density plasma is subject to microinstabilities (in most cases the ion acoustic instability) that are driven by the diamagnetic current*. The resulting anomalous field diffusion broadens the current sheath and strongly heats the electrons. Current understanding suggests that the piston instabilities will not be suppressed before the current sheath reaches a width $\approx c/\omega_{pi}$. Depending on whether the discharge tube radius R is larger or smaller than c/ω_{pi}, a different behaviour of the magnetic compression process is to be expected. In order that $c/\omega_{pi} < R$, the line density N in case of deuterium must be $N > N_{crit} = 3 \cdot 10^{15}\,cm^{-1}$ (that is, for R = 20 cm the critical density $n_{crit} = 2.5 \cdot 10^{12}\,cm^{-3}$).

Indeed, at densities below the critical density the plasma is swept up in the relatively broad current sheath (modified snow plough model [(7)] thereby mixing with the piston magnetic field. The compressed plasma therefore has a $\beta \lesssim 0,5$ [(4)(5)] (β is the ratio of gas kinetic pressure to pressure of the confining magnetic field). The ion temperature corresponds to a kinetic energy

*See also Section B.4(b) on Turbulent Heating as derived from the piston velocity in agreement with a picture of snow plough compression [5][6][7].

At higher densities (above n_{crit}) the current sheath is smaller being only some fraction of the tube radius. In this case the ions are elastically reflected by the piston (free particle model) and move ahead of it. This results in a $\beta \simeq 1$ plasma and in an ion temperature that corresponds to roughly twice the piston velocity (the electron temperature is usually much smaller) [3][7].

At densities of about the critical density, complicated radial plasma flow structures occur showing transitions from elastical reflection to snow plough behaviour during the plasma compression [8].

In the case of higher densities which is of great interest for high-ß toroidal confinement experiments (High-ß Stellarator, Belt Pinch) the following relation between ion temperature kT, density after compression n, compression ratio κ, and electrical field at theta pinch coil, E_Θ, holds (for 2 degrees of freedom)

$$kT \sim n^{-1/2} \cdot \kappa^3 \cdot E_\Theta .$$

For a step pulse of magnetic field the factor of proportionality is about 10^7 if one takes kT in keV, n in cm^{-3}, and E_Θ in kV/cm [9]. A plasma with kT = 15 keV and $n = 10^{15} cm^{-3}$ for example could in principle be obtained with $\kappa = 2$ and $E_\Theta = 6$ kV/cm, or alternatively with $\kappa = 5$ and $E_\Theta = 0.4$ kV/cm.

3. Applicability to large experiments and fusion reactors

In the high-voltage theta pinch experiments a technique has been developed (high-voltage low-energy capacitor banks [2][3][4] or pulse charged high-voltage transmission lines [1][5][6] that allows effective shock heating of low density plasmas to 10 keV temperatures at small compression ratios. With this method more than 10% of the stored energy is transfered to the plasma the remainder being mainly in the confining magnetic field.

Shock heating combined with subsequent adiabatic compression is the basis of a pulsed high-ß fusion reactor considered in [10].

Although the ultimate applicability of shock heating in fusion reactors is rendered difficult by the necessary large electric fields and relies on new concepts for the confining vessels it will play a decisive part in the next generation of large high-ß toroidal confinement experiments that are in the building or planning stage at the moment. This the more as shock heating with or without subsequent adiabatic compression is at the moment the only heating method that allows the production of a high-ß plasma close to fusion reactor conditions. The high-ß stellarators in particular, because of the small compression ratio needed for stability (wall stabilization), rely on shock heating unless some new yet unknown heating method can take its place.

Questionnaire (See pull-out supplement at end; extract from Section B.1)

1.) Yes, the required heating energy can in principle be obtained from fast high voltage condenser banks (bank energy of the order GJ). In order to get the necessary high field strength (a few kV/cm) at the circumference of the discharge vessel (E_{pol} in a torus), the shock heating coil can be subdivided into a number of sections, so that the capacitor bank voltage can be reduced to a convenient value.

2.) The electrical efficiency for energy storage in capacitor banks is almost 100%.

3.) The losses connected with the feed current (ohmic losses, inductive losses) in single pulse shock heating experiments can be kept low (about 10% of the stored energy after a quarter period, depending on the circuit components). If the shock heating coil has a higher temperature (e.g. in repetitively pulsed systems or for other reasons) the losses may considerably increase.

4.) Calculations for a pulsed staged theta pinch reactor [10] indicate, that the consequences of the current losses can be kept on an acceptable level.

5.) No gas load problem.

6.) No holes are required in a non-conducting wall; in a conducting wall toroidal gaps are necessary to avoid poloidal wall currents.

7.) No hardware is required within the vacuum vessel, if a non-conducting vessel or a metal liner of small thickness ($d \lesssim 0{,}2$ mm) could be used. Considerable difficulties are to be expected in fabricating (and cooling) a metallic liner with insulated gaps (for instance fabricated from a stack of aluminium washers with adjacent surfaces anodized for insulation), or shock compression coils insulated by a thin non-conducting layer (e.g. Al_2O_3 of 1 mm thickness), to avoid induced poloidal wall currents or breakdown across the coil gap, respectively.

8.) No problems with fast magnetic compression heating.

9.) More than 10% of the stored energy can be transferred to the plasma, the remainder being predominantly in the confining magnetic field.

10.) Strong shock heating with only weak plasma compression in the final state leads to high-ß plasmas; application of this heating method to low-ß experiments raises the problem of how the ß-value can be reduced. Shock heating in toroidal configurations requires means to avoid the toroidal drift to the outer wall (stellarator fields, toroidal currents plus copper shell or vertical fields).

11.) Fast magnetic compression heating leads to energies $E_i \gtrsim E_e$ and an anisotropic velocity distribution; only little is known about stability and plasma containment in the relaxation phase.

12.) 0.5 - 1 dollar/Joule (less than 10^{-3} dollar/kW, but no continuous heating).

13.) In principle to both linear and toroidal reactors. In a mirror reactor where continuous heating is necessary shock heating can only help to solve the problem of start up of such a reactor.

14.) At present fast magnetic compression is the only heating method that allows the production of plasmas close to fusion reactor conditions ($T_i \gtrsim 10$ keV, $n \approx 10^{13}-10^{14} cm^{-3}$). This heating method will certainly play a decisive part in the next generation of large high-ß toroidal confinement experiments (being built or planned e.g. in the laboratories at Culham, Garching, Jülich, Jutphaas, Los Alamos).

15.) The answer is no.

16.) No, it does not.

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Section B.4(d)

ADIABATIC COMPRESSION

Robinson, D.C., Kaufmann, M.[1], Bodin, H.A.B.

INTRODUCTION

This is not so much a heating method as a means of changing the state of a plasma so that possible limitations associated with other heating methods e.g. ohmic can be overcome and ignition possibly achieved.

Compression in a toroidal device can be envisaged as following an initial heating process such as shock heating or ohmic heating. To the initial heating might also be added neutral injection, high frequency heating etc before the slow compression is applied. Adiabatic heating is a part of the heating in nearly all pinch experiments, such as z-pinches, θ-pinches and toroidal configurations such as the screw-pinch, the M+S experiment and derivations of those configurations (reversed field pinch, belt pinch, high beta stellarator etc.). Usually after a fast non-adiabatic shock compression, which heats mainly the ions, the magnetic field increases further leading to an adiabatic compression. Three different situations can be distinguished. If the electron-ion relaxation time is small compared to the rise time of the magnetic field, both temperatures become equal and rise proportional to $B^{4/5}$ (if $\beta = 1$). For larger relaxation times the ion temperature stays above the electron temperature but both have the same B dependance. In high temperature and low density experiments the ion-ion relaxation becomes too small, hence one observes a two dimensional compression with $T_{i\perp} \sim B$. Experiments of this type reached temperatures in the region of 4 keV at densities of 10^{22} m^{-3} (1,2).

MINOR RADIUS COMPRESSION

The simplest means of applying an adiabatic compression is to raise the longitudinal stabilising field as performed on numerous high-β experiments and as envisaged in a pulsed high-β fusion reactor (3). On low β devices this is currently being studied on TUMAN (4) and in the near future on TOSCA (5). In these two devices an initial ohmically heated TOKAMAK plasma is subject to a rise in the stabilising field by a factor, C, of up to 4-5 in a time appreciably less than an energy confinement time. Assuming perfect conductivity during the compression then the nT

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product should rise by $C^{5/3}$ and the confinement time might be expected to improve because of better isolation from the external boundaries even though the plasma is smaller. Numerical studies of the changes in nT and τ_E for neoclassical plasma show somewhat more optimistic results[6,5] with nT increasing faster than C^2. The compressed plasma has a higher current density together with a reversed current outside it associated with an increased energy input from the circuit providing the current I. These two mechanisms lead to the additional increase in nT. This type of compression increases β_θ but not β, and q also increases. It is reasonable to ask whether the nT achieved using the higher ohmic heating current made possible at fixed q by using the final value of the longitudinal field exceeds that associated with compression or not. For pseudo-classical scaling only the optimistic results equal those to be obtained from the higher ohmic heating current. However in the quest for higher densities where radiation cooling is all important such a compression is very helpful. The Braginskii-Pease limit for ohmic heating against Bremsstrahlung cooling ($nT < \alpha J$ where J is the current density and α a constant) has already been reached in a number of present day devices due to their impure plasmas. In such a case the ratio of compressional heating to ohmic heating varies as $C^{2/3}$ which implies that the failure of ohmic heating to ignite a Tokamak plasma can be overcome by compression.

The biggest draw-back to the above method of compressing the plasma is that a large increase in total field energy is required. In an ignition experiment the main field coils will probably be operating close to their maximum field so that compression in the above way may not be practicable.

MINOR-MAJOR RADIUS COMPRESSION

By combining the radial compression with an inward compression of the plasma ring to a region of higher field using a controlling vertical field[7], it becomes possible to achieve a valuable increase in nT without any extra mainfield energy. For the <u>same</u> field volume nT increases approximately as $C^{5/3} \times \dfrac{1}{\left(1 - \frac{a}{R}\left(1 - \frac{1}{\sqrt{C}}\right)\right)^{10/3}}$

where a/R is the inverse aspect ratio of the torus. For a tight torus this can increase the nT product by a further factor of 1.5.

MAJOR RADIUS COMPRESSION

In this case the most valuable feature is that the main magnetic field can be left unchanged during the compression(7). The energy for compression is provided by the vertical field which is comparable to the plasma energy which in turn is much less than the total magnetic energy. The increase in nT is proportional to $C_1^{10/3}$ where C_1 is the ratio of the initial and final major radii.

Such a compression has been demonstrated very satisfactorily on the ATC device(8) where the adiabatic scaling laws have been verified (but not for the electrons) and the density has reached values in excess of 10^{20} m^{-3}. (A value which is normally associated with being above the radiation limit on TOKAMAKS.)

This type of compression principally raises β and the current must be increased by external means to satisfy flux conservation. Even though the total magnetic energy does not change during the compression the initially provided energy is large because of the large volume required to achieve a sizeable compression. If this volume had been used in the first phase to establish a larger current then on pseudo-classical scaling(9) there would have been no gain in nT using adiabatic compression. However as in the minor radius compression, if the plasma is radiation cooled then the ratio of compressional heating (nT) to ohmic heating in the full volume is $C_1^{4/3}$. This factor makes ohmic heating combined with compression in major radius at least a plausible means of attaining ignition.

ENERGETICS

The extra volume magnetic field energy needed to achieve the same increase in nT due to adiabatic compression by the three methods can be compared on the assumption (low β) that the extra energy needed to move the plasma and increase the current is a negligible fraction of the whole. For a tenfold increase in nT the optimistic minor-major radius compression uses less magnetic field energy than a full major radius compression but the reverse is true if a larger increase in nT is required. It should be borne in mind however that the minor-major radius compression only increases β slightly whereas the major radius compression increases β as $C_1^{4/3}$. The three types of compression are compared in Table I.

CONCLUSION

For low β toroidal reactors, adiabatic compression can provide a useful means of adding 100 MJ to the plasma energy. For slow ($\sim$ 100 msec) modest compressions with the energy coming from conventional flywheel generators (or a cryogenic magnetic storage system[3] which can recover much of the energy at the the end of the pulse[12] and using a minor-major radius or major radius compression the cost comes out to be for a present day room temperature system 2-3 \$/plasma joule. Using a low temperature inductive storage system this drops to 1 \$/plasma joule and if the plasma is re-expanded after the achievement of ignition by compression* - to prevent β_θ becoming too high due to α-particle heating, then the cost falls as low as 0.2 \$/plasma joule. It should be noted that the compression in the case of a TOKAMAK actually improves the stability properties of the system: however, results on present devices[4,8] show that the electrons are not heated adiabatically due to an anomalous electron loss process.

Even if adiabatic compression is easy to handle its applicability to a high β fusion reactor is limited. Strong adiabatic compression leads to a large volume filled with magnetic energy compared with a small plasma column, making such a reactor uneconomical. For special toroidal configurations, such as the high beta stellarator or the reverse field pinch, the adiabatic compression in addition suppresses the necessary stabilizing wall effect. Hence reactor concepts for high beta configurations include only a weak[3,10,12] or even no adiabatic compression[11].

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* An initially ohmic heated plasma at 10^{14} and 4 keV in the reference design which fails to ignite due to Bremsstrahlung when compressed to 3×10^{14} and 8 keV will yield 100 MW of α-particle heating. This will raise β_θ leading to equilibrium difficulties, consequently the most efficient use of the applied compressional energy is obtained by re-expanding the plasma in a controlled manner. For the minor-major radius expension, 70% of the applied energy would be recovered[3] reducing the cost to .3 \$/plasma joule. For the major radius expansion practically the full volume of the coils could be utilised reducing the cost in this case by about 5 x to 0.2 \$/plasma joule.

QUESTIONNAIRE (See pull-out supplement at end; extract from Section B.1)

1. Power available.
2. Flywheel generator (or cryogenic energy store?) to produce fields.
3. Leakage inductance and resistive losses are sizeable at room temperature.
4. Plasma unaffected by these losses.
5. No gas load.
6. No holes.
7. None.
8. Optimise type of compression.
9. Less than 50% in view of leakage inductance and eddy current losses.
10. Yes, but improves confinement and stability.
11. No change.
12. The basic cost of a flywheel generator delivering a slow compression pulse ($\gtrsim$ 100 msec) is 2.5 10^{-3} \$/joule (Siemens' quotation for a 6 GJ set). At such rates and at room temperatures only 25% is available inductively and hence cost is 10^{-2} \$/inductive joule. (If capacitor banks had to be used (pulse rise $\lesssim$ 1 msec) then the cost would be 0.4\$/inductive joule.) The major cost of each of the compression schemes is the effective extra toroidal field energy required. For the major radius compression this has to rise by 4 to 6 times depending on aspect ratio for a 100 MJ increase in plasma energy (reference design). For an initial β value of 2% this gives a cost of 2-3\$/plasma joule, with a similar figure for other types of compression. Utilising re-expansion of the plasma and low temperature coils this figure falls 0.2 \$/plasma joule.
13. Toroidal and linear reactors.
14. Does not solve the Problem.
15. Can modify the plasma current distribution in a favourable manner but not maintain it.

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TABLE I

Type of Compression	a	R	Plasma Energy	B^2	Field Volume	Current Increase	Vertical Field Increase	Control Field Energy Increase
Minor radius	$C^{-1/2}$	const.	$C^{2/3}$	C^2	const.	const.	g (c)	< 30%
Minor-major radius	$\sim C^{-1/2}$	$\sim \left(1-\frac{a}{R}\left(1-\frac{1}{\sqrt{c}}\right)\right)$	$\sim \frac{C^{2/3}}{\left(1-a/R\left(1-\frac{1}{\sqrt{c}}\right)\right)^{4/3}}$	C^2	const.	$\sim\left(1-\frac{a}{R}\left(1-\frac{1}{\sqrt{c}}\right)\right)^{-1}$	g (c)	$\sim 2\left(1-\frac{a}{R}\left(1-\frac{1}{\sqrt{c}}\right)\right)^{-2}$
Major radius	$C^{-1/4}$	$C^{-1/2}$	$C^{2/3}$	const	f (c)	$C^{1/2}$	$\gtrsim C$	$\sim 2C$

$$g\,(c) = \frac{B_v{}^f}{B_v{}^i} = \frac{I^f R^i}{I^i R^f}\,\frac{(\ln\, 8R^f/a^f + \beta\theta^f - 1.25)}{(\ln\, 8R^i/a^i + \beta\theta^i - 1.25)}$$

$$f\,(c) = \frac{\left[\left(1+\frac{1}{\sqrt{c}}\right) + \frac{R}{a}\left(1-\frac{1}{c}\right)\right]^2}{2\left[\left(1+\frac{1}{c}\right) + \frac{a}{R}\left(1-\frac{1}{\sqrt{c}}\right)\right]}$$

TRANSIT TIME MAGNETIC PUMPING

by

M. Brambilla
in consultation with W. Millar, C.N. Lashmore-Davies and A. Messiaen

1. THEORY

Collisionless Transit Time Magnetic Pumping (TTMP) is based on the Cerenkov absorption of energy by the plasma from a low amplitude oscillating electromagnetic field. Efficient absorption in a low β plasma(1) implies $\omega/k_{\parallel} \simeq v_{\theta i} \ll v_A \ll c$ ($v_{\theta i}$ and v_A being respectively the ion thermal velocity and Alfven velocity). Moreover, obviously, $k_{\parallel} \sim 1/L$ where L is a typical dimension of the device; and since $L \gg \rho_i$ (the average ion Larmor radius) it follows that $\omega \ll \omega_{ci}$ (the ion cyclotron frequency). Under these conditions the compressional and torsional Alfven waves reduce essentially to two evanescent vacuum waves, with

$$k^2 \simeq - k_{\parallel}^2 \left(1 + 0 \left(\frac{v_i^2}{v_A^2}\right)\right). \tag{1}$$

The resonance of the torsional Alfven wave appears to be accessible however for the plasma parameters of a reactor(2),(3).

The dielectric properties of the plasma only appear in the coupling of these externally excited waves with the electrostatic ion sound wave. In turn, this electrostatic field contributes efficiently to the power absorption.

The linear theory of TTMP heating of a Maxwellian plasma is well developed(4)-(7). In these works a uniform cylindrical plasma column was considered, acted upon by a compressional wave (with $\tilde{B}_1 \parallel B_o$). More recently the possibility of using a torsional wave in the case of a toroidal plasma was suggested by F. Koechlin and A. Samain(8). Toroidal corrections were included in a more general way by E. Canobbio(9),(10). In all cases the power absorbed per unit volume can be written

$$P_{LIN} = n\kappa T \frac{\omega^2}{k_{\parallel} v_{\theta i}} b^2 \exp\left(- \frac{\omega^2}{k_{\parallel}^2 v_i^2}\right) Q\left(\frac{T_e}{T_i}, \frac{\omega}{k_{\parallel} v_{\theta i}}\right) \tag{2}$$

where $b = B_1/B_o$ is the modulation rate, and Q is a dimensionless function of order unity which was extensively tabulated by E. Canobbio and S. Giuffre(11).

The limits of validity of the linear theory are also well understood. E. Canobbio(9),(10) pointed out that the ion collision frequency must exceed a critical value for the linear theory to be valid. If this condition is not satisfied, a local "plateau" develops in the interaction region in velocity space, and the power absorption is correspondingly reduced. In the general case one can write

$$P = P_{LIN}/(1 + n_c/n) \qquad (3)$$

where n is the actual density, and

$$n_c = \frac{b^{\frac{3}{2}} \; T_i^2}{R \; \ln \Lambda} D \left(\frac{\omega}{kv_{\theta i}}, \frac{T_i}{T_e}\right) \qquad (4)$$

(R is the major radius of the torus, $\ln\Lambda$ the Coulomb logarithm). If T_i is in eV and R in cm, D varies between 10^{13} and a few times 10^{14} in the interesting region, and was also tabulated in reference(11).

Due to the simplicity of the theoretical model it was also possible to study some of the effects of TTMP on the plasma confinement. The most obvious of these effects could be the disruption of the magnetic surfaces. For real Stellarator fields (Wendelstein(12) and Proto-Cleo(13)) numerical calculations have shown that this only occurs at very large modulation rates (greater than 10%). For Tokamaks similar calculations have not been made. The required modulation rate in the case of a reactor is however so small (a few parts in a thousand) that one can reasonably rule out such a possibility.

In connection with a negative experiment on Proto-Cleo(14) M. Brambilla(15) has pointed out that electrostatic fields associated with TTMP can give rise to a particle flux from the outer plasma layers by essentially the "neoclassical" mechanism involving the resonant particles. By a proper design of the coils and by the use of electrostatic screenings, this effect should however be small enough not to affect the plasma confinement time.

It follows from its "resonant" nature that TTMP is a "slow" heating method, in the sense that the heating time can be made much longer than the ion collision time. From Eq.(2) it also follows that power absorption is uniformly distributed within the plasma column, so that no important modification of the temperature and density profiles has to be expected. Thus one can exclude any important perturbation from the local thermal equilibrium which could drive instabilities in the plasma.

In conclusion, as far as the present status of the theory is concerned, no appreciable deterioration of the plasma confinement time should be inherently related to the application of TTMP.

2. APPLICATION

If we compare the power and energy requirements for the ignition of a low-β toroidal plasma with the possibility of generating this power, we find that TTMP is at least presently in a favourable position. Power sources in the TTMP frequency range would require a relatively modest development to match the requirements of a reactor, and the cost per watt should be comparable or even slightly lower than for other HF heating schemes.

From Eq.(2) one finds that a modulation rate of one part in a thousand (130 G oscillating magnetic field) could typically be needed. For a set of meridional one-turn coils of 2 m radius at about the Helmoltz position the required current is of the order of 50 kA.

The ohmic losses in the coils have been evaluated by Wort(17), Koechlin and Samain(2), and by Cato et al(18). The geometry considered in each case varies largely; all these calculations however agree on the conclusion that ohmic losses will not exceed a few MW per coil.

Losses in the reactor walls should also be acceptably small provided some space is available between these and the coils.

Cato et al(18) have also considered in some detail the problem of removing the ohmic heat (as well as the heat due to neutron bombardment) from the coils. These authors point out that as long as a coil structure has to be used, TTMP poses less severe cooling problems than heating at higher frequencies. This is because at lower frequencies the skin depth is larger and the heat deposition in the coil more uniform, so that the danger of surface melting is reduced. They conclude that for the above figures it should be possible to evacuate the Joule heat by heat-pipe techniques with sufficient speed to allow steady operation.

On the other hand, because of obvious penetration problems, the coils cannot be separated from the plasma by any closed metallic surface. This raises a host of technological problems to which only very preliminary, if any, solutions have been proposed.

The problem of radiation damage of the coils is common to all techniques requiring metallic structures inside the blanket; its solution has to be assumed for the realization of the reactor independently of the heating method chosen. The problem of feeding the high frequency power through the thick blanket has already been mentioned in reference(18). While probably not insoluble, no detailed proposal for its solution is yet available. The most formidable problem specific to TTMP, however, is probably the insulation of the HF voltage. For the numerical values mentioned above the peak HF voltage at the coil terminals is as high as 100 kV. Breakdown problems in the residual gas (at an expected pressure of about 10^{-4} torr) will be very severe, especially if this gas is partially ionized, as stressed in reference (18). To improve the situation one could envisage(19) the use of structures of high dielectric constant (such as loaded alumina), or azimuthal insulation of the blanket cells with corresponding repartition of the HF voltage between the sectors.

T. Consoli[20] suggested a tubular metallic structure as a liner which could be made highly transparent to the HF field, but opaque to charged particles leaving the plasma. Such a structure could ease the task of insulating the coil terminals and reduce the contamination of the plasma; at the same time it could play the role of electrostatic screen. However the supporting, cooling and radiation damages of such a structure are still open questions.

It is worth mentioning that the possibility of using a version of TTMP to help in the solution of the refuelling problem was considered theoretically by A. Samain[21]. On the other hand the use of a travelling TTMP wave for seeding the Tokamak equilibrium current does not look feasible on energetic grounds.

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NATURAL RESONANCE HEATING

by

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S. Puri and M. Tutter (Sections 3 and 4)

A. Messiaen and P. Vandenplas (Section 2.4)

1. INTRODUCTION

This section deals with RF heating excluding ion TTMP. It includes heating at a frequency in the ion cyclotron frequency range, the lower hybrid frequency range and the range of the electronic frequencies ω_{pe}, ω_{ce}, ω_{UH}, etc. For fusion plasmas, there is little hope that the latter can be used, because the necessary power supplies will hardly be available. Accordingly, we will restrict ourselves to the first two frequency ranges mentioned above.

2. HEATING BY AN ELECTROMAGNETIC WAVE IN THE ION CYCLOTRON FREQUENCY RANGE.

2.1 Generation of the pump into the plasma

The heating energy is transferred into the plasma by a compression wave at a frequency ω smaller than the lower hybrid frequency at the plasma edge. The heating takes place because of a direct resonance of the pump with the particles, when $\omega = \omega_{ci}$ or $\omega = 2\omega_{ci}$. The heating may be due also to a non linear effect(1), when $\omega > 2\omega_{ci}$: the pump then drives unstable plasma electrostatic modes which in turn resonantly transfer their energy to the particles. Finally the heating may also be due to a resonance with electrons of TTMP type, if $\omega \sim k_{\parallel} V_{th_e}$.

The pump may be an eigen-mode of the cavity consisting of the liner filled with the plasma. The effect of such a resonance which may increase the heating efficiency has been studied by the Brussels Group in the case of various structures and frequencies of the pump. Experimental evidence for the compression wave resonance exists (see e.g. reference (2)). The pump is generally induced into the cavity by a coil system, but it could be induced through wave guides: (see Fig.1). This last technique seems the most promising. With a magnetic field B = 50 kG and D^+ions, such wave guides should have a minimum size $\frac{\lambda}{2} = \frac{\pi c}{\omega} = 2$ m, if $\omega = 2\omega_{ci}$, and $\frac{\lambda}{2} = 1$ m, if $\omega = 4\omega_{ci}$ and could eventually be used in a full size reactor. As suggested by J. Adam these dimensions could be reduced by a factor ~ 3 if the wave guides are filled by alumina. In the simplest situation, the pump is symmetric around the magnetic axis. The electric field then consists of an inductive part E_θ and of a self-consistant electrostatic part E_r. The magnetic perturbation

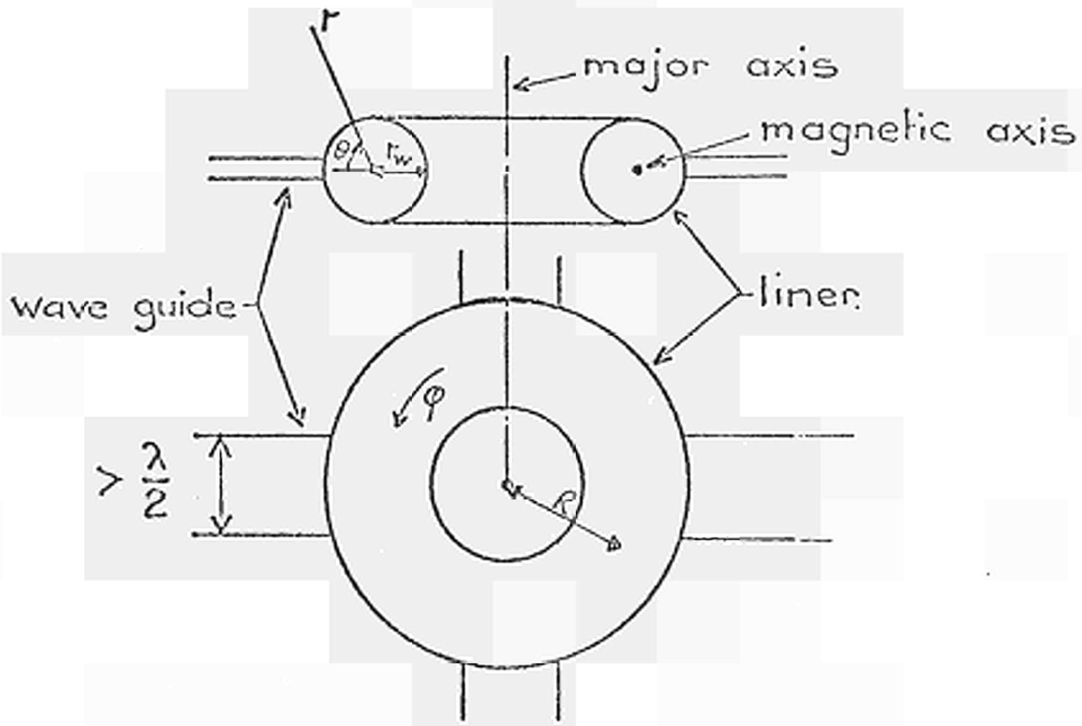

Fig.1 Approximate layout of a waveguide feed system

$\tilde{B}$ is mainly in the ϕ direction. If the number 2N of guides is large, the mode $\sim \exp iN\phi$ is preferentially excited. The phase velocity along r is $\frac{\omega}{k_r} \approx C_A$ where C_A is the Alfven velocity. Assuming first that there is no power absorbed in the plasma or in the liner, the fields E_θ and $\tilde{B}$ in the plasma vary roughly as:

$$\left.\begin{aligned} E_\theta &\approx X \left(\frac{C_A}{r}\right)^{\frac{1}{2}} \sin\left[\int_0^r k_r dr\right] \\ \tilde{B} &\approx Xc \frac{1}{(r\, C_A)^{\frac{1}{2}}} \cos\left[\int_0^r k_r dr\right] \end{aligned}\right\} \quad r > \frac{1}{k_r} \tag{1}$$

where X is a constant. The mode is an eigen-mode if:

$$\int_0^{r_w} k_r dr = \nu\pi \quad \nu = 1, 2, \ldots \tag{2}$$

For a plasma density $n = 10^{14}$ cm^{-3} (D^+ ions), a magnetic field $B = 5.10^4 G$, and a liner radius $r_w = 100$ cm, the value of ν for $\omega = 2\omega_{ci}$ is ~ 10. Due to the variation of C_A with r, E_θ does not increase very much from the plasma edge to the centre. However the focusing effect associated with the $r^{-\frac{1}{2}}$ dependence of E_θ may be useful if non linear heating is used (in which case a large value of E_θ is necessary in a relatively small region).

When there is a small power absorption, the resonance of the pump into the liner has a meaning if the damping rate defined as

$$\gamma = \frac{W_L + W_P}{\mathcal{E}}$$

(where W_P and W_L are the power absorbed by the plasma and the liner and

$\mathcal{E} \sim [\frac{\hat{B}^2}{4\pi} \times \text{volume}]$ is the energy associated with the pump) is less than the frequency shift $\Delta\omega$ obtained by changing ν into $\nu + 1$ in (2):

$$\gamma \ll \Delta\omega \sim \frac{\omega}{\nu}.$$

This condition is satisfied in the practical cases considered below. A point of importance is the value of the electric field E_g at the junction between the wave guides and the liner, which must be as small as possible. This value is effectively minimum when the condition (2) is satisfied. In that case E_g is in phase with $\hat{B}$. Given the power W_P into the plasma, the factor X which appears in (1) is determined. The value B_w of $\hat{B}$ at $r = r_w$ and the power W_L are also determined. The field E_g is then determined by:

$$W_L + W_P \approx S\, c\, E_g\, B_w \frac{1}{4\pi}$$

where S is the total cross section of the guides. The electric field into the guides is determined by the boundary conditions $E = E_g$ and $\hat{B} = B_w$ at the guide junction.

2.2 Heating at $\omega = \omega_{ci}$ and $\omega = 2\omega_{ci}$

When $\omega = \omega_{ci}$, the field E_r is such that the circular polarization $E_r + iE_\theta$, which rotates in the direction of the ion cyclotron motion, is nearly cancelled. Heating is possible only if this screening effect is not total. The best scheme seems to make the pump resonant with a low density assembly of special ions mixed with the bulk of the plasma ions[(3)] (e.g. H^+ ions with a density 10^{-2} - 10^{-3} times the density of D^+ and T^+). The pump gives energy to this assembly, which in turn heats up the plasma. The heating power which is finally achievable is at the best of the same order found in the case where $\omega = 2\omega_{ci}$. However, heating at $\omega = \omega_{ci}$ has been successfully used in several experiments (see e.g. references (2) and (4)).

When $\omega = 2\omega_{ci}$, the average power density which can be delivered to a Tokamak plasma is given by[(3)]

$$W = \frac{nT}{\tau} = n \frac{1}{20} (K_\perp^2 \rho_{th_i}^2)\, \omega_{ci} \frac{R}{a} m_i \left|\frac{CE_\theta}{B}\right|^2 \quad \text{(gauss units)}$$

where $K_\perp = \omega/C_A$, R and a are the major and minor radius of the plasma, n and T are the plasma density and temperature, m_i is the ion mass the ρ_{th_i} is the ion thermal Larmor radius. For a given ion species the heating time varies as $\tau \propto \frac{B^3}{n E_\theta^2} \frac{a}{R}$. For $n = 10^{14}$, $B = 50$ kG, $\frac{R}{a} = 5$ and D^+ ions we obtain

$$\tau = \frac{nT}{W} = 300 \text{ ms}$$

for $E_\theta = 30$ V/cm. With R = 5m and a = 1m the power W_p to the plasma is ~ 10 MW. This power may be supplied by alumina filled wave guides with a total cross section $S \sim 2.10^4 \text{cm}^2$, the electric field in the guides being ~ 2000 V/cm.

Experimental evidence of heating at $\omega = 2\omega_{ci}$ may be found, e.g. in reference (6).

2.3 Heating at $\omega > 2\omega_{ci}$

When $\omega > 2\omega_{ci}$, heating by direct resonance of the pump with the particles is impossible. One must turn to heating through electrostatic modes. The possible candidates are the ion Bernstein modes, the sound waves and the drift modes. They may be driven unstable either by parametric effects or by non-linear Landau effects (in the first case the pump decays into two electrostatic modes, in the second the pump decays into one mode and one quantum of parallel or transverse particle energy). The calculated threshold of the pump electric field E_θ for the onset of these instabilities is very small in the case of a uniform magnetic field. In a realistic configuration, however, the non-linear resonances which are necessary for the pump decay take place in a small domain of space. If one takes into account this effect and the finite group velocity of the electrostatic modes, one finds much larger threshold values of E_θ. Detailed calculations are being done by the Julich group. These calculations are more optimistic than a calculation by N. Martinov and A. Samain in a Tokamak, which gives[5]

$$\frac{E_\theta \text{threshold}}{B} = 20\ \frac{V_{th_i}}{c}\ \frac{\omega_{ci}}{\omega_{pi}}\ \left(\frac{\rho_{th_i}}{R}\right)^{\frac{1}{2}}\quad \text{(gauss units)}$$

where $\omega_{pi}^2 = \frac{4\pi m c^2}{n_i}$ and $V_{th_i} = \omega_{ci}\,\rho_{th_i}$. If $n = 5.10^{14}$; $T = 4$ keV (D^+ ions), $B = 50$ kG, $R = 5$m, then $E_{\theta thr} = 200$ V/cm. One must notice that even if the threshold values of E_θ could be practically achievable, it might well happen that the unstable electrostatic modes could not carry the heating power from the pump to the plasma. Assuming that the total cross section of the guides is 10^4 cm^2, a power of 10 MW may be supplied to the plasma with an electric field into the guides ~ 1000 V/cm.

In the heating schemes described above the energy is supplied to ions in the transverse direction. The heating rate must be smaller than the collision rate for the anisotropy to remain small. This condition is satisfied in the Tokamak case.

2.4. TTMP of the electrons

If $\omega \sim |k_{\parallel}|v_{th_e}$, the exciting signal gives to the electrons a power density of

$$W = \sqrt{\pi}\, NT\omega\, (\tilde{B}/B)^2$$

with a condition on the power density similar to the condition for ion TTMP. This heating method is interesting because the value of $\tilde{B}$ inside the plasma, for a given exciting external current, can be strongly enhanced by a magnetoacoustic bounded plasma resonance if the exciting frequency is well chosen[14].

The magnetoacoustic resonances can be divided into two types (1) the standing compressional Alfvén wave resonances which occur for $\omega < |k_{\parallel}| V_A$ when $k_{\perp} a \cong \chi_{n,i}$ ($\chi_{n,i}$ being the i^{th} root of Bessel function of order n for an $e^{in\theta}$ azimuthal field dependence and $k_{\perp}^2 + k_{\parallel}^2 = \omega^2/V_A^2$ if $\omega^2 \ll \omega_{ci}^2$, V_A is the Alfvén velocity). (2) A torsional Alfvén wave resonance appearing for $\omega \simeq |k_{\parallel}| V_A$ when $|k_{\parallel}| V_A \ll \omega_{ci}$ and shifting towards ω_{ci} when $|k_{\parallel}| V_A$ increases.

The major advantages of this heating scheme are: (1) The occurrence of the resonance corresponds to resonance of the whole system constituted by the bounded plasma - toroidal cavity system. This will help to solve the coupling problem because of the, at least partial, cancellation of the reactance seen by the coupling system. (2) For a given value of $\tilde{B}$ and $k_{\parallel}$, the wall losses are multiplied by a factor $\sqrt{V_{th_e}/V_{th_i}}$ with respect to ion TTMP while the dissipated power in the plasma increases as D V_{th_e}/V_{th_i} with respect to ion TTMP, D being a factor depending on the field amplification due to the Q of the toroidal cavity at resonance.

The transfer of energy from the electrons to the ions will occur through energy equipartition; this requires that the energy equipartition time be sufficiently small with respect to the energy confinement time which is a condition fulfilled in large machines.

It is interesting to note that heating (multiplying the temperature by two) has already been achieved on present generation tokamaks without loss of stability[15]. The absorption mechanism in these experiments is non linear parametric instability, the existence of this mechanism being, however, difficult to assess in reactor conditions. It should be underlined that such a mechanism is not needed since the method would rely, in reactor conditions, on electron TTMP for the absorption of energy.

2.5 Particle diffusion

The diffusion of the particles under the influence of the pump or the electrostatic modes induced by the pump may be shown to be small in the following intuitive manner (which may be used also in the case of heating at lower hybrid frequency). When a wave with a frequency ω and a transverse wave number $k_\perp$ is absorbed by particles, it transfers a momentum $\Delta p_\perp$ and an energy ΔE related as

$$\Delta p_\perp = \frac{k_\perp}{\omega} \Delta E \tag{3}$$

the momentum transfer $\Delta p_\perp$ for a particle corresponds to a transverse displacement across the magnetic field $\Delta r \sim \frac{c}{eB} \Delta p_\perp$. If the particles absorb a power $m V_{th}^2/\tau$ per particle, they experience a transverse drift which is given by

$$V_d \sim \frac{c}{eB} \frac{k_\perp}{\omega} \frac{m V_{th}^2}{\tau} .$$

In the presence of a series of modes, the particles experience drifts from each mode in different directions. The resultant drift therefore verifies

$$V_d \lesssim \frac{c}{eB} \frac{k_\perp}{\omega} \frac{m V_{th}^2}{\tau}$$

It follows that the life time τ_p associated with the presence of the modes or the pump verifies

$$\tau_p^{-1} \sim \frac{V_d}{r} \leqslant \frac{c}{eB} \frac{k_\perp}{\omega} \frac{m V_{th}^2}{\tau}$$

i.e.

$$\tau_p \gtrsim \frac{r}{k_\perp \rho_{th}} \left(\frac{V_{th}}{\omega}\right)^{-1} \tau .$$

With $k_\perp \rho_{th} \lesssim 1$ and $\omega \sim \omega_{ci}$, $\tau_p \lesssim \tau \frac{\rho_{th}}{r} \ll \tau$. This conclusion may be deduced more rigorously from quasi-linear theory. It may be noticed that a relation similar to (3) applies as well to the parallel momentum transfer to the particles from the pump on the magnetic axis

$$\frac{dp_\parallel}{dt} \sim \frac{k_\parallel}{\omega} \frac{dE}{dt}.$$

This mechanism cannot be used to maintain the current in a Tokamak, however, because it would need too large a power. To have $dp_\parallel/dt$ (per cm^3) of the order of $ne\,\eta\, I$ would need a power $\frac{dE}{dt}$ (per cm^3) $\sim \frac{\omega}{k_\parallel} ne\,\eta\, I \sim (\frac{\omega}{k_\parallel V})\, \eta I^2 \gg \eta I^2$ (η is the plasma resistivity, $I = ne\overline{V}$ is the electrical current density).

3. HEATING AT THE LOWER HYBRID FREQUENCY

3.1 Principle

This method consists of launching an electromagnetic pump with a frequency ω equal to the lower hybrid frequency ω_{LH} inside the plasma. The transverse wave number and the amplitude of this wave reach large values in the resonance region $\omega_{LH} \approx \omega$. In collisional plasmas this gives as a result a direct thermalization of the pump. In hot plasmas the pump is preferably converted first into electrostatic modes, which propagate until they are damped by resonance with the particles. The conversion may be either linear or non linear. In the first case the pump experiences a change of its structure according to its full dispersion relation. In the second case, its amplitude reaches the threshold value for the onset of the instability of electrostatic modes. There is considerable amount of theoretical and experimental work on the conversion mechanisms(7) showing that the energy of the pump would be efficiently thermalized into the plasma.

3.2 The accessibility problem

In the usual confined plasmas $\omega_{LH} = \omega_{pi} = (4\pi m e^2/m_i)^{\frac{1}{2}}$. The vacuum wave length for $n = 10^{14}$ and D^+ ions is $\lambda = 20$ cm. The pump must have a finite wave number $K_\parallel$ along the flux lines for having access to the resonance region without first crossing a region where it becomes evanescent. The necessary value of $K_\parallel$ is $\sim \frac{\omega}{c}$ and it might be provided by a slowing down structure or simply by the finite size of the wave guide carrying the pump into the liner. It is not obvious however that a good efficiency of the transmission of the pump from this wave guide to the plasma interior can be easily achieved.

The problem is that the electromagnetic wave launched by the antenna, during its advance into the magnetized plasma of gradually increasing density, converts into a slow electrostatic wave, eventually encountering a region of very high refractive index and low group velocity in the vicinity of the lower hybrid resonance. Before the wave reaches the hybrid layer, it is in general obliged to tunnel through a region of evanescence where the wave energy may be reflected. Various authors(8)-(10) have pointed out that this problem could be avoided if the refractive index n_z along the magnetic field direction is made large enough, the required value being invariably larger than unity. Such a solution necessarily entails the usage of either coils or slow-wave structures within the reactor walls, which is undesirable even if not altogether ruled out.

In a recent study[11], the authors have shown that for the plasma and magnetic field parameters of a thermonuclear plasma, the evanescent region near the plasma edge is thin enough to allow effective tunnelling even for $n_z < 1$. The immediate impact of this result is that one can dispense with all structures within the reactor walls. The maximum plasma density accessible using this scheme is given approximately by the relation $\omega_{pe}/\omega_{ce} \sim 0.2$ or

$$n_{max} = 4 \times 10^3 \times B^2(g)\ cm^{-3}.$$

For B = 100 kG

$$n_{max} = 4 \times 10^{13}\ cm^{-3}.$$

The lower-hybrid-frequency for this case is about 0.8 GHz.

3.3 Coupling

For the case $n_z \lesssim 1$, the most optimum coupling occurs for TM modes incident at grazing angles on the plasma surface. Therefore, the waveguide antenna should be oriented so as to launch TEM like modes on the plasma surface as shown in Fig.2

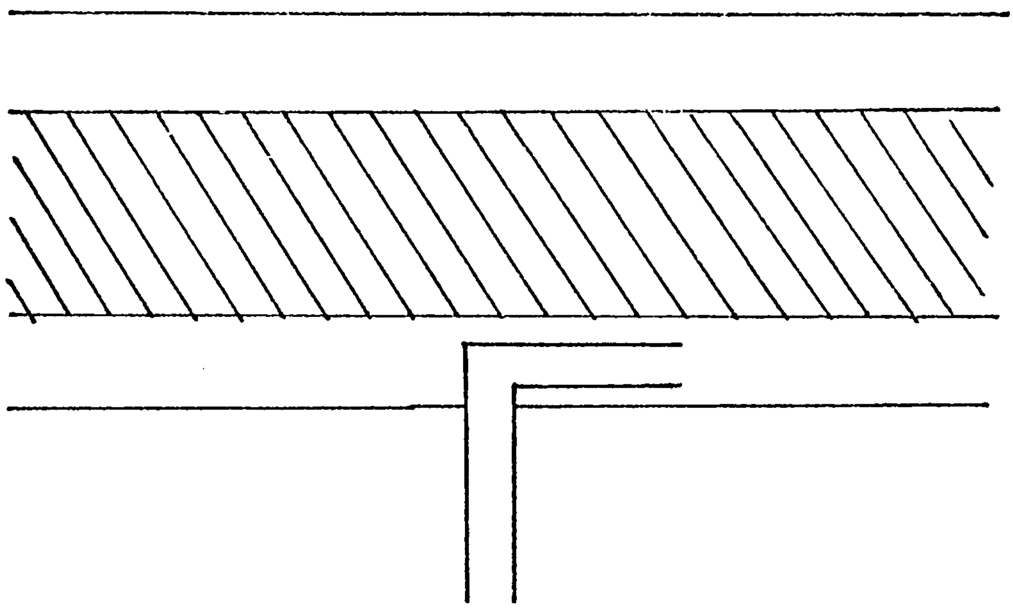

Fig.2 Orientation of wave guide to launch TEM waves.

It is difficult to evaluate the coupling efficiency (fraction of input power entering the plasma). Instead we make a pessimistic estimate assuming that the antenna possesses no directivity whatsoever. In this case all angles of incidence are equally probable. From Fig.3 we note that approximately 1.5^{o} integrated aperture is available where all the power enters into the plasma. This aperture is insensitive to the plasma radius and may be assumed to exist even in a thermonuclear plasma. A simple calculation shows that in such a case, the probability that the RF energy enters the plasma is approximately 10^{-4} and is the same as for stainless steel (or niobium). The resultant coupling efficiency is 50%. The fact that the antenna does indeed possess directivity will further enhance the coupling and may lead to coupling efficiency of about 90%.

3.4 Matching

Since the waveguide is easily matched to the vacuum, it is possible to radiate all the energy into the reactor volume. Due to the broad antenna pattern of the waveguide, this energy spreads itself both longitudinally and azimuthally, so that the probability of significant reflection back into the waveguide due to subsequent scattering from obstacles is negligible. Thus the generator will stay matched to the load even if parameter changes occur in the plasma.

Ultimately this may prove to be the decisive advantage of lower-hybrid-heating over other RF heating schemes.

3.5 Remarks

In the cold-plasma approximation the energy absorption takes place through electron-ion collisions and the energy is principally absorbed by the electrons. In what manner the energy will be absorbed and what precise role the lower-hybrid resonance will play in a hot plasma is not clear. Also what possible problems could arise from the extremely thin absorption region is far from obvious[12][13].

It is difficult to give precise cost estimates at present but the figure 0.2 UC/watt can be taken as a guideline. (As an example: 20 tubes Varian VKP-8259 could generate 12 MW of 500 MHz during 1s with an efficiency of about 50% for $1.2 million, i.e. 0.1 UC/J).

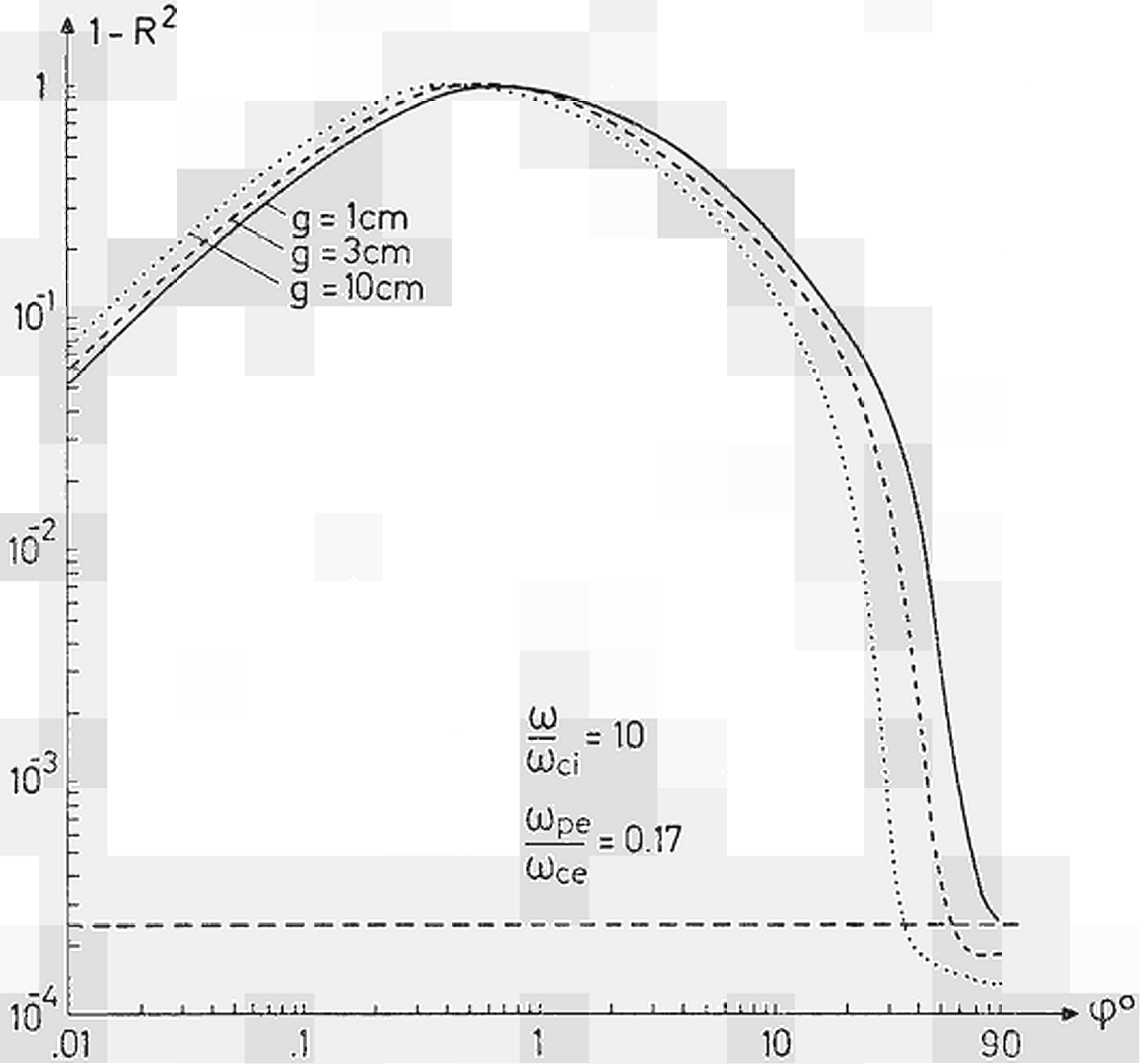

Fig.3 Power transmission coefficient as a function of angle of incidence.

4. CONCLUSION

Efficient RF heating is possible at $\omega = 2\omega_{ci}$ and $\omega = \omega_{LH}$, and is perhaps possible at $\omega_{LH} \gg \omega > 2\omega_{ci}$. The necessary power supplies are technologically feasible. There is no theoretically predicted degradation of plasma confinement. As usual with RF heating, one major problem is to induce the pump in the plasma. The heating operation will be much simpler if the wave can be launched into the plasma through waveguides. This is perhaps possible at $\omega = 2\omega_{ci}$ in a full-size reactor, and in that case, heating at $\omega = 2\omega_{ci}$ would appear an excellent method. It is probably possible at $2\omega_{ci} < \omega < \omega_{LH}$. Heating at $\omega = \omega_{LH}$ may also be possible, using waveguides to launch TEM modes, although if $B \lesssim 100$ kG this approach appears to have an upper limit to the plasma density of around 4.10^{13} cm^{-3}. At higher densities, some slow wave structure is probably required, and in this case, breakdown or matching problems may arise.

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LASER PLASMA HEATING

I.J. Spalding (Culham)

INTRODUCTION

1. The inertially-confined laser-compression (LC) approach to fusion is growing very rapidly, with support in the USA which now approaches magnetic confinement budgets[1]. However, the laser may also find a less glamorous, but technologically less exacting, role in the following areas:

1.1 Ignition of steady-state toroidal reactors[2].

1.2 Ignition of quasi-steady-state reactors.

1.3 Generation of localized plasmas -
eg. for build-up in mirror systems[3]
or manipulation of current skins in toroidal systems[4].

1.4 Pellet refuelling -
eg. acceleration into, or rapid dispersion of the pellet upon arrival at, the major axis.

1.5 Preionization or preheating of plasmas prior to adiabatic compression[5] - to decouple the reactor wall design from technological problems which may be encountered with alternative (high voltage) preheating techniques.

1.6 Parametric control -
eg. to reach density or temperature regimes otherwise unattainable by joule heating, or to avoid the external generation of gross magnetic fields etc.

2. The potential field of application is thus very wide. Since laser technology is still advancing very rapidly, it is not yet possible to provide a definitive survey of the field - of the type one might attempt for joule heating, for example. To illustrate the potential advantages and disadvantages of laser heating, we will therefore first consider the questionnaire detailed in Sect. B.1 in a general way, and then discuss next-generation experiments and a few specific longer-term suggestions in greater detail.

3. QUESTIONNAIRE* - Relevance to large experiments and fusion reactors

3.1 POWER:- the requirements for toroidal reactors are of the order 10 - 100 MW. (Sect.B.2). Continuous laser powers > 60 kW have been produced and discussed in the literature; the upper limit is not

*(See pull-out supplement at end; extract from Section B.1)

yet established.[19] 2kJ (20μs) pulsed lasers have been described, and 10kJ (0.1 - 1ns) lasers are now being designed for LC experiments[1]; it seems reasonable to extrapolate these energies to ~ 50kJ for longer-pulse operation, and to assume that convectively-cooled fluid laser media could operate at repetition rates of 100 - 200Hz. (These rates have already been achieved at lower energies). Thus, mean powers of 10MW per unit are conceivable, assuming that suitable mirror and window techniques can be scaled to these higher powers. (An array of ~ 10 units around the major axis of the reactor might then permit operation at ~ 100MW.)

3.2 EFFICIENCY:- the following electrical efficiencies have been reported:

Pulse duration	Efficiency (η)	Laser Medium
(a) Continuous	~ 50%	CO (λ = 5μm)
(b) ~ 10^{-5}sec.	~ 25%	CO_2 (λ = 10μm)
(c) ~ 10^{-7}sec.	~ 10%	CO_2 "
(d) ~ 10^{-9}sec.	2 - 3%	CO_2 "

Chemically-pumped lasers have operated at $\lambda \sim 3\mu m$ with electrical efficiencies > 100%. The quantum efficiency of the laser represents an <u>upper limit</u> to total energy efficiency, and can be some 40% (CO_2) to 90% (Xe_2^*). We may conclude that η is unlikely to be high enough for heating mirror reactors[6] but that ignition of α-heated steady-state stellarators should be possible (subject to §3.9 and §3.12), and may be conceivable for pulsed Tokamaks, etc,

3.3 TRANSIT LOSSES:- these could be $\lesssim$ 10%.

3.4 CONSEQUENCE OF LOSSES:- transit and target reflection losses due to stimulated Brillouin back-scatter present heating and isolation problems to the laser designer; wall problems due to classical reflection/refraction should be avoidable.

3.5 GAS LOAD:- possibly. (Imperfect ionization, or $\beta > 1$ expansion across the confining field.)

3.6 ACCESS HOLES:- these should be compatible with the neutronics. (Saturation, or breakdown flux in the final amplifier or window may imply laser beam diameters of order 100cm; however, the convergence of the focussed beam is a design variable, so that access ports of only 10 - 30 cm diameter in the reactor blanket may be feasible.)

3.7 HARDWARE:- photons (unlike neutrons, neutral atoms or charged particles) can be specularly reflected and focussed into the vacuum chamber, irrespective of the electric and magnetic field configuration, and with no internal hardware. This may constitute a significant practical advantage for laser heating.

3.8 UNIFORMITY OF HEATING:- homogeneous heating is virtually impossible - conduction or convection is necessary to ensure that power permeates the whole plasma.

3.9 COUPLING EFFICIENCY:- overdense plasmas ($\omega_{pe} > \omega$) have been heated in LC experiments with very high efficiencies. The loss is due to reflection/scattering, and can be < 1%[7]. In under-dense plasmas the efficiency may be much lower, cf § 5 , and Refs. 2, 5, 9 - 16.

3.10 ELECTRIC AND MAGNETIC FIELD:- asymmetric irradiation, or $\beta \geqslant 1$ expansion, will produce strong, local, transient, perturbations of the magnetic field.

3.11 DISTRIBUTION FUNCTION:- varies with the scheme involved; counter-streaming normally expected.

3.12 CAPITAL COSTS:- a laser capacitor cost of 10^{-4}uc/W(E) generated was estimated for a laser-stellarator ignition scheme[2]. This cost is small compared to that of the associated fly-wheel generator and charging supplies, which seem likely to dominate other component and laser assembly costs. Neglecting the refuelling components, and assuming charging, capacitor, and energy-supply costs of 2×10^{-2}uc/W, 2×10^{-1}uc/J, and 5×10^{-3}uc/J respectively, the total heating cost is of order 0.025 uc/W(E) generated (ie. 0.37 uc/watt deliveried to the plasma), if one assumes an overall heating efficiency of 10%.

3.13 REACTOR-CONCEPTS AMENABLE TO LASER-HEATING:- toroidal reactors and linear θ-pinches.

3.14 NEXT-GENERATION EXPERIMENTS:- proposals are being prepared (and discussed with other Euratom laboratories) by Garching and Culham, (§4).

3.15 REFUELLING:- it does not solve the problem, but may help pellet-injection schemes.

3.16 PLASMA CURRENT?:- no.

NEXT-GENERATION EXPERIMENTS

4. Two ad-hoc working parties representing interested Euratom laboratories have met at Garching on 4 June and 13 July 1973; the next meeting will be convened by Spalding, at Culham, in the autumn. Table 1 summarizes Euratom toroidal experiments in which laser-heating or filling is currently being considered; the second column lists the earliest date at which the laser techniques might be applied. Garching plans to apply laser-filling techniques[20] to W IIB and W VII, and - in conjunction with Grenoble -

to WEGA. Culham intend to fill Cleo-Stellarator in a similar manner, and may later apply similar techniques in DITE. Liquid hydrogen droplets[21] are the preferred target, and λ = 10.6μm is the preferred wavelength. Although Frascati have no experimental commitment at present, they have considered:

i. Laser-pellet preionization for F.T. large Tokamak.

ii. Local preionization in F.T., to manipulate current skins[4].

iii. Preionization and heating to obtain a collisionless banana-regime in TTF II[8].

REACTORS

Absorption

5. For a thermonuclear temperature of 10keV and plasma densities n = 10^{17}, 10^{18}, 10^{19}/cc, the required confining field is B $\geqslant$ 0.28 MG, 0.9 MG and 2.6 MG respectively. Only the first of these is within the capabilities of steady magnetic field technology, and classical heating processes - inverse Bremsstrahlung and Synchroton absorption - are extremely weak for these values of n and B[5]. However, oscillating two-stream and ion-acoustic decay instabilities occurring near the critical density, and the 2 ω_{pe} decay instability occurring near one-fourth the critical density can strongly enhance the light absorption, so that λ = 10.6μm radiation should strongly heat plasmas having densities in the range 2.5 × 10^{18}/cc to the critical density at 10^{19}/cc. (The corresponding densities for λ = 1μm radiation are 2.5 × 10^{20} - 10^{21}/cc.) Note that stimulated Raman (SRS) and Brillouin (SBS) scattering in under-dense plasma may "anomalously" enhance reflection; an additional self-focussing instability at ~ $2\omega_{pe}$ may lead to filamentation of the light as it propagates into the plasma.

6. Thus, strong "anomalous" heating of near-critical (ω/ω_{pe} = 1 or 2) plasmas can be expected above the threshold intensities discussed, for example, in References 9 and 10. (The effects of plasma gradients, the spatial, temporal and spectral structure of the laser beam, and also its polarization and angular convergence when focussed, must be considered in quantitative discussions).

7. The following, weaker, heating effects may also be important in under-dense plasmas:

i. Kinetic instabilities[11]

ii. Stimulated Compton scattering[12 - 15]

iii. Two lasers - beating at ω_{pe}[16], but detailed reactor assessments (or confirmatory experiments) have not yet been attempted for these effects.

LASER-AUGMENTED θ-PINCH REACTORS

8. Dawson et al.[5] have investigated the possibility of using CO_2 lasers to heat a magnetically confined (ie. under-dense) θ-pinch plasma by inverse bremsstrahlung. This concept requires that a hollow density distribution, established with reverse-trapped fields, should act as an efficient plasma light-pipe over distances of some hundreds of metres, or that stable self-focussing should be established (with minimal SBS back-scatter, which is tilted by the plasma gradients), over comparable axial distances in a plasma approximately one ion gyro-radius wide. A variant discussed by Vlases (private communication) is a laser-heated core to a magnetically-insulated gas-blanket. Each of these approaches avoids the high coil voltages which might prove a technological embarrassment to "shock" heated reactor concepts, but they encounter other physical (eg. conduction) and economic/technical problems encountered in linear θ-pinch reactor assessments[17, 18].

LASER-IGNITED PELLET-REFUELLED α-HEATED TORUS

9. CLM R109[2] takes advantage of the high densities necessarily generated during pellet-refuelling of a steady-state toroidal reactor to invoke anomalous CO_2 heating at the critical density, so that efficient single-pass heating of the pellets should be possible. It is assumed that the $\beta > 1$ expansion phase does not significantly perturb the confinement, and illustrative parameters are then discussed[2] for a Stellarator refuelled by pellets of a few mm diameter. This technique might also be used to augment ohmic, or other heating methods, (provided that the ablation problem can be solved with realistic pellet dimensions and velocities). If pellet-refuelling proved to be technologically impossible in a full-scale reactor plasma, the "moving limiter" concept might still permit low-density/small radius start-up, with the focal zone of the laser moving in a programmed manner from the centre to the outer periphery of the plasma column. In this application, there is a less severe constraint on the minimum pellet dimension, so that the laser repetition rate can be higher (eg 100 - 1000Hz); however, the dimension of the access port would need to be greater (along the minor radius) and, of course, some other means of refuelling would be needed after ignition.

CONCLUSIONS

10. Kilojoule laser pulses of 10^{-8} - 10^{-7} second duration are now technologically possible, and are of interest for heating next-generation Euratom toroidal plasmas. Garching and Culham are planning to extend their present experimental programmes to energies of interest for these experiments. (The Lebedev has a modest toroidal-filling programme, whilst United Aircraft are planning to use laser-plasmas as charge-exchange targets for 10kV H^0 injection in "reactor-like" mirror experiments[(3)]).

11. These experiments, coupled with strong international efforts on laser development for laser-compression schemes, should permit a more informed assessment of the potential of laser-heating schemes for magnetically-confined reactors: our provisional assessment should be B/C for reactor applications.

25 October 1973

Table 1 (Compiled by Dr S. Witkowski, IPP, June 4, 1973)

	LPP - EXP; Date	Volume (cm^3)	Density (cm^{-3})	max. Number of Part.	Temp. (eV)	Pumping speed (l/sec)	Size of Ports vert. (Ø mm)	horiz.	Wall Radius internal/ external/ outside B-field coils	B_{max} (kG)	Confin. Time (msec)
CLEO	1975	$2\cdot10^5$	10^{14}	$2\cdot10^{19}$	100-1000	300	70	70	14/20/50	20	10
DITE	1976 ?	$8\cdot10^5$	10^{14}	$8\cdot10^{19}$	100-1000	>> 300	?	?	25/28/70	28 - 33	?
FT	1976 ?	$7\cdot10^5$	$10^{13}-10^{14}$	$7\cdot10^{19}$	10 - 100	?	400 x 30	400 x 30	25/- /70	100	> 100
TTF	1975 ?	10^5	$10^{13}-10^{14}$	10^{19}	500	?	?	?	~15/ ?	20 (?)	1-10 ?
W IIb	1974 -75	$3\cdot10^4$	$5\cdot10^{12}-10^{14}$	$3\cdot10^{18}$	100-1000	200	30	50	9/13/20	15	1-5
WEGA	1974/75	$2\cdot10^5$	$5\cdot10^{12}-10^{14}$	$2\cdot10^{19}$	100-1000	> 1000	50	100	22/- /50	15/25	10-20
W VII	1976/77	$5\cdot10^6$	$5\cdot10^{12}$-1014	$5\cdot10^{20}$	100-1000	10^5	100	200	35/ - /110	40	100
Mirror											

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NEUTRAL INJECTION METHODS

J G CORDEY, R A DEI-CAS, J P GIRARD, A C RIVIERE, G SCHILLING, J SHEFFIELD

1. DEFINITIONS

1.1 The term 'Neutral Injection' applies to several processes and purposes. Generally a particle is introduced into the plasma by this method and it is originally 'neutral' simply to allow penetration of the magnetic field surrounding the plasma. There are three main definitions of the purpose of neutral injection:

(i) HEATING. The temperature of the plasma is raised by injecting ions at an energy E_o substantially greater than the plasma temperature T and most of this energy is transferred to the plasma by collisions resulting in an increase in T.

(ii) MAINTAINING EQUILIBRIUM. The equilibrium of the plasma is modified by the particle distributions resulting from neutral injection in a desirable way such as the production of a plasma current to maintain a Tokamak discharge.

(iii) REFUELLING. The loss of deuterium ions by fusion reactions and diffusion must be compensated by the injection of ions with E_o much lower than T for toroidal systems and with E_o approximately equal to T for open-ended systems.

These applications will be examined in this order in terms of the questions suggested by C J H Watson in Section B.1 and as applied to the proposed model reactors of types 1, 2 and 4 but not 3 (shock and compression heated torus).

2. HEATING

2.1 Of the three model reactors, heating by neutral injection is considered with respect to types 1 (Stellarator) and 2 (Tokamak), although many of the conclusions also apply to type 4 (mirror reactor).

Probable Requirements

2.2 The minimum power required to heat a Tokamak reactor similar to model 2 in addition to that obtained from ohmic heating was given by Sweetman[1] as approximately 30 MW. The actual power required will be reduced by the gain from nuclear reactions between the beam and plasma in the manner described by Artsimovich[2]. This will amount to 20 per cent at the most. To keep the heating time short (a few seconds) it may be necessary to increase this and for the present discussion the power required will be taken to be 100 MW, a value also used earlier by Sweetman et al[3] for similar reactor models. The Stellarator reactor has similar dimensions and containment time and again 100 MW is taken as the power required. This power could be divided between 10 injectors supplying 10 MW each.

2.3. Penetration to the axis of the plasma by the beam in the perpendicular direction would require a beam energy of about 1 MeV[4] for a mean plasma density of 3×10^{14} cm^{-3} and in the tangential direction about three times this energy. The energy required for penetration varies as $n_e^{1.23}$ so that at 10^{14} cm^{-3} the energy for perpendicular injection is only 260 keV. At 1 MeV each injector must supply a neutral beam intensity equivalent to about 10 amperes. Recent calculations by Girard et al[5] suggest a reduction in the energy required to the 0.1 MeV range by the use of moving limiters. Alternatively, if heating of the outer edge of the plasma is permitted then again energies in the 0.1 MeV range could be used. The injectors must then supply 100 amperes each or the number of injectors must be increased. Mauriette and Girard[26] have pointed out that the mean free path for ionization may be reduced by impurities (approximately $\sim (Z_{eff})^{-1}$).

2.4 The efficiency of the system for producing a neutral beam of required power can in principle be greater than 90 per cent but in practice may be somewhat less[10]. Most of the beam power is obtained in the main acceleration stage for the ions and by careful restriction of the beam emittance prior to acceleration this process should be loss-free. The efficiency of neutralization[3] can be high if direct recovery is applied to the un-neutralized portion of the beam. Without this energy recovery the efficiency for conversion of a molecular ion beam is greater than 20 per cent and for a negative ion beam greater than 80 per cent. For present experiments the overall efficiency can be quite low, i.e. less than 50 per cent [11][22].

2.5 The efficiency of transfer of the beam from neutraliser to plasma can be very high in principle with losses mainly arising due to collisions with background gas. With a flight path of 6 metres for example, an average gas pressure of 10^{-5} torr and collision cross-section at the most of 10^{-16} cm^2 then about 2×10^{-2} of the beam is lost. This represents 2 MW of beam striking the flight tube walls at a power density of less than 3 Watt/cm^2. The beam tube through the blanket has a finite pumping speed for gas evolved in the tube and this imposes a limit on the current. A simple criteria for this limit is that the total current should be less than $\frac{e\,S}{\sigma L \gamma}\left(1 - \frac{n_o \sigma L}{f}\right)$ where

- e is the electronic charge
- σ the cross section for electron loss by beam atoms
- L the tube length
- γ the wall re-emission coefficient
- S the pumping speed (approx. 100 a^3/L litres/sec)

and f the acceptable level of attenuation of the beam. For example with $\sigma = 10^{-16} cm^2$ (for 500 keV beam energy and a mixture of air and hydrogen), $L = 200$ cm, $\gamma = 1$, tube radius $a = 20$ cm and f of 2 per cent, this limiting current is 22.4 amperes. This rather severe limit simply implies that the tube must be made of a large enough radius to allow escape of the gas and the wall re-emission coefficient must be kept small. The increase in tube size need not be large since $S \alpha a^3$.

2.6 The slow gas current associated with the beam must be kept small compared with the beam level, which means less than 10^{19} sec^{-1} per injector. This represents a contribution of only 0.03 per cent of the total plasma content appearing locally on the outer surface. More information is required about the acceptable level of cold gas input to the plasma before a practical limit can be set.

2.7 For each of the 10 injectors the hole in the reactor wall need only be of the order of 40 cm in diameter, the total for all being less than 0.2 per cent of the reactor wall area. For neutral injection it can also be said that no hardware will be required inside the vacuum vessel. The distribution of the power in the plasma could be controlled by arranging the injection directions appropriately and adjusting the intensity and energy for each injector.

2.8 In general an injected ion will travel many times around the torus in a slowing down time. Experimentally, slowing down of fast ions consistent with classical friction has been observed[22]. A small loss of the injected ions will occur by charge exchange near the plasma edge and by injection into orbits reaching the vacuum wall. A detailed study of the particle distribution in a reactor model may be necessary to help determine the advantages and disadvantages of perpendicular over tangential injection.

Present State of the Art

2.9 Existing ion sources[4][5][6][22][23] are operating with extracted current densities in the range $0.25 \rightarrow 0.5$ A/cm^2 in hydrogen and deuterium for energies $15 \rightarrow 30$ keV and total extracted ion current ~ 10 amperes. This performance is within a factor of (2) of the theoretical perveance limit. The beam divergence is typically ± 2 degrees. Thus the plasma sources are already operating at a level consistent with the above requirements (2.3) (2.4) (2.7), but we will need post acceleration to achieve the required beam energies. In addition effort is required on the extraction electrode design to increase the beam pulse length from the present $0.01 \rightarrow 0.1$ sec to the required 10 sec level.

2.10 Heating of the Tokamak plasma by neutral injection has now been studied in several experiments[22][24][25] at beam powers up to 70 kW, energies up to 25 keV and pulse lengths of 25 mseconds. There were no disastrous effects on the plasma stability or confinement and typically the ion temperature was raised by 20 per cent with in one case a large increase in neutron output[24]. This increase in T_i was consistent with theoretical estimates[24]. Although precise figures are not known, only about 25 to 50 per cent of the beam powers available were transferred into the plasma. These losses occured along the flight path and also at the plasma because a proportion of ions were injected into uncontained orbits.

Costs

2.11 The cost of high-voltage, high-power systems for neutral injectors was estimated by Julian[3] in 1970. The cost for ten injectors giving 10A each at 1 MeV (D^o) was £11M and this would be approximately £15M (35 MUC) at present-day prices. This was an extrapolation based on commercial rectifier-convertors operating at up to 600 kV and included 2 spare supply systems, switch gear and control equipment. The cost of the remainder of the injector equipment (without direct conversion of the unwanted high-energy ion beam) would be expected to be a smaller figure, so that total cost ~ 50 MUC.

Disruptive Effects

2.12 Several possible disruptive effects could occur with neutral injection. Early theoretical work on neutral injection into toroidal plasmas suggested that there would be considerable changes in the equilibrium of the background plasma and possible enhanced plasma loss. The current state of the theory on these disruptive effects is as follows:

(a) Toroidal momentum balance with parallel injection

It was first shown by Callen and Clark[12] from consideration of the toroidal momentum balance equation, that with one beam and injection parallel to the magnetic axis of the machine the background plasma would rotate with a very high velocity in the toroidal direction (for the present and future generation of experiments the toroidal velocity was greater than the sound speed). This problem could be somewhat reduced by using another neutral beam of the same current injected in the opposite direction; however complete cancellation of the injected toroidal momentum cannot be achieved everywhere.

Recently it has been shown by Connor and Cordey[13] that small ripples in the toroidal field due to finite coil spacing can destroy the toroidal momentum of the background plasma. Calculations for the present and next generation machines indicate that the toroidal velocity will be quite small ($U_\varphi \sim 0.1\ C_s$) and the equilibrium should be only slightly perturbed.

(b) Poloidal electric fields in the case of perpendicular injection

This problem has recently been discussed by Callen and Clark[14]. Briefly these authors conclude that with perpendicular injection, since most of the hot ions will be trapped in 'banana orbits', this will induce a poloidal field E_θ which could then convect the plasma across magnetic surfaces ($\underline{E}_\theta \wedge B_T$ drift).

This calculation is not a self-consistent equilibrium calculation and contradicts the numerical work of Bishop and Smith[15] on toroidal equilibria which suggested that poloidal electric fields due to ion banana trapping would rapidly decay.

Also E_θ does not give net particle drift in the banana regime since the azimuthal momentum is conserved. Some losses may occur if E_θ is large enough for the drift paths to intersect the wall. This part of the injection theory may require further consideration.

(c) Microinstabilities of the energetic ions

Unless the hot ions are injected isotropically which would mean several sources then the hot ion velocity distribution may be unstable both during the initial switch-on phase and when the distribution has reached its equilibrium configuration. This topic has been discussed by Cordey and Houghton[16] and Stix[17]; conditions are given for the stability of the ion distribution in these papers. It is found that electron Landau damping stabilizes most of the modes (ion acoustic, drift etc.) in the case of parallel injection; however for perpendicular injection several modes are unstable. Although these microinstabilities may affect the heating rates of the ions and electrons, they should not perturb the equilibrium significantly.

3. MAINTAINING TOKAMAK EQUILIBRIUM

It was first shown by Ohkawa[18] that the injected ions can give rise to a significant current in the plasma. There have been several theoretical calculations of this current[18][14][16] although it has not been observed experimentally yet. Ohkawa[18] showed how the current could be made large enough to run a Tokamak in a steady state mode. If one takes into account the bootstrap effect[19] then a smaller injection current will suffice[20]. This would seem to be a very important field for research since the establishment of a steady state current would improve the reactor prospects of the Tokamak significantly.

4. REFUELLING

4.1 Mirror Reactors are refuelled continuously by injection of ions at an energy close to the mean plasma temperature. The power requirements have been discussed in the Report of the Advisory Group on Open Adiabatic Configurations and Radio Frequency Plugging[10] and are about an order of magnitude greater than required for heating a toroidal reactor with the additional requirement that the system must have very high power efficiency. The injection energies are lower (~ 500 keV) and would be slightly easier to attain than the $1 \rightarrow 3$ MeV required for toroidal systems. For mirror machines as envisaged at this time neutral injection is essential so that the existence of a mirror reactor programme implies a considerable development programme on high current ion sources, possibly for negative ions.

4.2 Toroidal Reactor. The refuelling of a toroidal reactor by neutral atom injection would seem to require very low atom energies in order to keep the circulating power in the station to a small fraction of the output power. A simple relation for the fraction of the total electrical power required for injection can be derived[21], namely

$$F = \left[\eta_g \; (1 + Q \, \eta_i \; f/2E) \right]^{-1}$$

where

η_g = generator efficiency = 0.5

η_i = injector efficiency = 0.8

Q = fusion energy = 20 MeV

E = injection energy

f = fractional burn-up of fuel = 0.05 .

For $F = 0.1$, 0.2, 0.3 and 1.0, E must be 21 keV, 44 keV, 70 keV and 400 keV respectively so that for sensible circulating powers E must not be greater than a few 10's of keV and very much less than the energy required for penetration.

If it is essential to penetrate to the centre of the plasma then single atom injection is not a method for refuelling a toroidal reactor of the type considered here.

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PLASMA HEATING BY CLUSTER-INJECTION

by R Klingelhofer*

with contributions from F Bottiglioni[+], J Coutant[+], M Fois[+], W Henkes*

As space charge is the physical barrier which limits the flux density of charged particle beams, cluster ions with their relative low charge-to-mass ratio are expected to be an appropriate way to diminish this difficulty.

A second property of clusters is their broad mass distribution due to their statistical growth process. Since all clusters have the same energy after being singly ionized and acceleratedthe mass-distribution is transformed into a velocity-distribution. This in turn should be useful to counteract the growth of plasma instabilities.[(1)]

Production of the cluster beam

A hydrogen cluster beam is produced by expansion of gaseous hydrogen out of a cooled nozzle. During the expansion the jet cools further down and at sufficient high gas density condensation occurs, transforming a certain amount of the gas jet into a cluster beam. The uncondensed gas is removed effectively by fractional pumping. After passing two skimmers a pure cluster beam enters a high vacuum chamber.[(2)]

Due to the method of production the cluster size is not at all uniform but has a rather broad distribution function the width of which is comparable to the most probable size. The mean size increases with increasing pressure of the gas upstream the nozzle orifice, with increasing nozzle diameter and decreasing temperature.[(3)(4)] Mean sizes between a few molecules and $\approx 10^9$ have been produced up to now.[(5)]

Ionization and acceleration

After leaving the beam producing system, the clusters are ionized by electron impact and subsequently accelerated by electrostatic means. In present experiments accelerating voltages are of about 1 MV which corresponds to a mean energy of 10 keV per atom for a mean size of 100 atoms per cluster ion.

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In principle, it is possible to increase the energy by one order of magnitude (up to 10 MV, 100 keV per atom); cluster ions can experience very high electric fields without being destroyed; low current 10 MV electrostatic accelerators have already been built and a 2.5 MV, 200 mA cascade has been constructed and could probably be developed to 5 MV without major difficulty.

Other accelerating processes have also been suggested, e.g. continuous electron ring acceleration(6) by travelling waves, but at the present time none has been tested.

State of the art

Clusterion accelerators of 600 and 1000 kV are operated by Association EURATOM-CEA sur la Fusion Contrôlée at Fontenay-aux-Roses (France)(7) and the Gesellschaft fur Kernforschung, Karlsruhe (Germany) respectively achieving cluster beam currents between 30 and 70 mA equivalent. A 1 MV, 10 A equivalent *clusterion accelerator is under construction at Karlsruhe with the final goal to be installed at the Wendelstein VII stellarator in Garching (Germany).

At one time, it seemed to be difficult to produce clusterion beams with a mean size of the clusterions of about 100 hydrogen atoms per clusterion at the current level in question because the cluster size and the beam intensity increase simultaneously and cannot be adjusted independently. More recently it has been shown that the mean value of the mass distribution of the neutral beam can be shifted to lower values in the ionized beam by the electron bombardment in the ionizer, so this problem has been overcome.(4)(7)(8)

Neutral hydrogen cluster beams with 10 A equivalent and a mean size of ≈30,000 hydrogen atoms per cluster have already been produced, as well as a several hundreds of A equivalent at a mean size of about 10^9 hydrogen atoms.(5)

Questionnaire+ of the Advisory Group on Heating and Injection

1. The required heating power for large experiments such as JET is not yet available but credible.
2. Overall efficiency of a system producing accelerated cluster ion beams has not yet been measured. However, it is possible to evaluate some orders of magnitude. Most of losses are produced in the accelerating stage by beam

*1 Ampere-equivalent = current of 6.10^{18} atoms/sec

+See pull-out supplement at end; extract from Section B.1

divergence and clusterion neutralization (charge exchange on residual gas). Neutral cluster beams can be produced with a divergence less than 1^o. A clusterion beam, with mass number of 100, carrying 10 A equivalent current, should not diverge (by space charge effects) more than 1^o. Proton beam-charge exchange losses can be reduced down to less than 0.3% in an accelerating tube with working pressure of about 10^{-8} mmHg. This estimation is made with pessimistic assumptions of a charge exchange cross-section equal to cluster geometrical cross-section, and an accelerating stage 5 m long. Therefore we believe that a 10 A equivalent clusterion beam can be created with efficiencies greater than 40%-50%. The large gyration radius of clusterions makes them less sensitive than protons to magnetic field. As an example, cluster ions with mass 100 accelerated to 1 MV or 10 MV have a Larmor radius of respectively 28 cm and 90 cm in a magnetic field of 5 Wb/m^2. However it is still necessary to neutralise the clusterion before injection into a reactor. This process could be made with much higher efficiency than for an ion beam, if neutralization cross sections are confirmed to be proportional to cluster geometrical cross sections, as appears from some preliminary measurements.

3. See point 2.

4. Consequences of losses are not known yet. In the example of point 2, efficiency 50% and power lost of a few MW, the mean power on walls of accelerating stage would be of about 10 W/cm^2, which is acceptable if tube is externally cooled by high pressure gas.

5. There is no gas load problem as far as the production of the accelerated cluster beam is concerned. The gas load due to the neutralizer which may be necessary, is expected to be smaller than in the case of atomic or molecular ions.

6. The inherent quality of the low charge-to-mass ratio should reduce the necessary size of the hole diameters compared to those for atomic beams.

7. There will be no hardware within the vacuum-chamber.

8. As the ionization cross sections of the clusters are known[8], the power is expected to permeate the plasma of a reactor or correspondingly large machines only in the case of low density start up at a level of about 10^{13} cm^{-3} and further α particle heating is possible if the energy per atom of cluster is 10 keV. If the energy per atom is raised up to

100 keV, cluster could be used in large machines at density levels of 10^{14} cm^{-3}. It must be pointed out that there is an increase of 75% in penetration, with respect to a neutral atom beam, computed for a joint clusterion beam and neutral atom injection.[10]

9. As mentioned under 8. the total absorption of the power is not considered to be a problem.

10. There are no fields affecting particle confinement

11. The beam will have a broad velocity distribution, which seems to be favourable as far as plasma instabilities are concerned.

12. In so far as powers and voltages of power supplies are comparable with what is contemplated for neutral beam injection ($V \approx 1$ - 3 MV, $W \approx 10$ MWatts) the cost can be guessed to be about 0.2 UC/Watt. We believe, however, that such a price refers to standard production and not to prototypes. For 10 MV power supplies initial cost is very difficult to estimate. For prototype power supplies, costs will be higher than 0.2 UC/Watt.

13. The method may be applicable for a toroidal reactor in case of low or intermediate density ignition as mentioned under point 8. Cluster ions with 100 keV/atom could be injected into open systems also.[9]

14. The method will be tested on the Wendelstein VII-stellarator.

15. Clusters may also be useful for the refuelling of plasmas. For this purpose they should be large enough that the electrons of ionized atoms in the cluster cannot leave it unless the ionizing event occurs in a surface layer, the thickness of which is of the order of a few atomic distances in the condensed phase. If this is the case the ionization rate of the clusters is reduced and the penetration length increased. As an upper limit the diameter of the clusters should not exceed the absorption length of the radiation which is emitted by the recombination of the electrons after ionization. In this case only a very small part of the energy of impinging particles will be absorbed and ablation should be greatly reduced. A rough guess is that the cluster size should not exceed $\sim 10^8$ hydrogen atoms per cluster. Clusters of this size have already been produced[5]. Provided that they can be ionized to the theoretical limit given by ion field emission, cluster velocities of the order of 10^4 m/s could be obtained using a 1 MV accelerator. Other methods of acceleration may be possible to obtain higher speeds, but speeds in excess of 10^5 m/s are undesirable for refuelling.

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GUN PLASMA INJECTION

H.C. COLE and W. BIEGER

Plasma guns have been used to provide plasma for many fusion experiments in both closed and open systems, although with limited efficiencies of perhaps up to 10%. On a reactor scale their possible use can be considered under four headings.

(1) The provision of plasma at thermonuclear energy

(2) The provision of a relatively cool plasma to be heated by some other method.

(3) Refuelling

(4) To supply the primary ions for a neutral beam injection system.

The first three items involve the direct injection of plasma into the containment region of the reactor. The injection problem is therefore discussed first in general terms.

The Injection problem

If we assume that we have a stream of plasma to be injected into a reactor system, the first limitation results from a consideration of Liouville's theorem. This indicates that the density of the contained plasma cannot be significantly larger than that in the incoming plasma stream unless the latter is strongly localized in velocity space. This is strictly valid only for a collisionless plasma, but can be regarded as an approximation in this case.

It follows from this that the energy flux per unit area in the stream must be comparable with the random energy flux in the plasma, i.e. $\sim 10^8$watts/cm^2 over an area of a few square centimetres, and maintained for times of a few seconds. A possible injector for the first category must therefore be capable of continuous operation delivering 10 keV plasma at $\sim$10kA/cm^2 or able to be pulsed at such a rate as to maintain this mean flux.

There are basically two methods of injecting plasma into a containment system.

(1) Across lines of force

(2) Along lines of force.

The first method requires the plasma to be transported to the containment region, and hence some form of guide field is required. One could possibly envisage this to be combined with the divertor system. Assuming that we deliver the required stream of plasma to the separatrix it will polarize in the magnetic field. If the plasma is to propagate across the field lines this polarizing field must be maintained. However, the helical form of the containment field will tend to short out this polarizing field when one pitch length in the field system becomes populated with plasma.

In addition the polarizing field produced in the injection region may greatly increase pump-out during the injection time once the plasma has populated the outer region of the containment system. This is due to the outward force generated by the polarizing field acting over that part of the plasma periphery not occupied by the injector beam itself. The situation is further worsened by the fact that populated field lines are equi-potential and hence the polarizing field will extend over a much greater width than the injected beam so greatly increasing the loss area.

It appears therefore that cross-field injection is not a viable proposition at reactor level.

Turning now to injection along field lines we consider first the closed line system. It will be necessary to extract the containment field in order to inject plasma onto it, and then return it to its equilibrium position before all the plasma escapes again. This means the whole operation must take place in a time less than the transit time of plasma round the torus. For the reactors we consider this is $\sim 25\mu$ sec; a daunting problem.

It is possible to pull out part of the field, or even more simply to inject through the divertor. In either case one only populates an outer layer, and the time restriction still stands.

In the case of open ended systems plasma can be injected along field lines without greatly perturbing the system. The injector would then be situated in the mirror or in a guide field with the same or slightly greater field strength than the mirror.

This necessitates the injector working in a magnetic field of > 100 kG.

The injection requirements for a mirror reactor are stringent. If we inject at 500 keV with a beam density equal to the plasma density in the reactor then the equivalent current density in the beam will be > 30 kA cm^{-2}. As the total current requirement is ~ 3 kA eq. the beam diameter would need to be ~ 4mm. Plasma sources with such characteristics have not yet been developed.

<u>Requirements for Toroidal Reactors</u>

1. <u>Start up plasma</u>

   This needs a device to supply ~ 2 X 10^{22} particles at ~ 10 keV in times ~ 1 sec.

   i.e. ~ 100 MJ minimum

   Current density ~ 10 kA cm^{-2}

2. <u>Cool Plasma</u>

   Supply ~ 2 X 10^{22} particles in times ~ 10 sec

   Injection at say 1 kA cm^{-2} at 100 eV.

3. <u>Refuelling</u>

   Supply ~ 10^{22} particles s^{-1} can be cold in principle but the penetration problem raises a serious difficulty.

4. <u>see Neutral Injection Paper</u>

   Primary ion energies of 1 - 3 MeV required to build ~ 10 MW 1 MeV neutral injection units.

Parameters of Plasma Guns

	Beam density n cm^{-3}	Energy eV	N pulsed	N sec^{-1} CW	Efficiency
Thetatron	10^{15}	500	10^{17}-10^{18}		~ 5%
Bostick	10^{16}	200	10^{17}-10^{18}		~ 5%
Coaxial	--	30	10^{15}		
RF Accelerators	$\sim10^{7}$	10^{4}		$\sim10^{18}$	10%
Crossed Field Injectors		30	10^{15}		5%
Hall Accelerators	10^{12} 10^{9}	10^{3} 10^{4}	$\sim10^{19}$	$\sim10^{22}$ $\sim10^{18}$	~50% ~
Compressed Flow Devices	5×10^{17} 1-7×10^{15} 1-5×10^{17} 10^{17}	4-6 800 1200 100	4×10^{21}		50%

Discussion

The arguments presented on the question of direct plasma injection suggest that it is not a viable proposition at reactor level.

In addition the characteristics of known plasma guns and accelerators do not match the requirements for supplying start-up plasma for a reactor.

The only device which has some possibility of providing the required plasma beam is the so-called compressed flow device. Even this requires particle energy to be increased by $\sim 10^2$ and the time scale by $\sim 10^4$, but the particle flux would need to be reduced by $\sim 10^2$.

As the beam density in these devices is $\sim 10^{17}$ the possibility of plasma injection at $\beta = 1$ may be worth considering.

With regard to circumventing the consequences of Liouville's theorem by subsequent compression of the plasma one might consider the possibility of injecting plasma during the time at which the magnetic field is rising. This may be the only reasonable solution to the injection problem.

From the point of view of supplying cool plasma and for refuelling the quantities of plasma could be supplied by either a compressed flow device, a Hall accelerator or possibly an RF accelerator but the injection problem remains.

Assuming that plasma guns could be used either to fill or startup a reactor, only a very approximate figure can be given for the cost. On the basis of 20% efficiency one would require 5 MJ for filling, cost ~ 1 MUC, and 500 MJ for startup at a cost of ~ 100 MUC.

With regard to supplying the primary ions for a neutral beam injector, none of the accelerators would appear to be capable of development into the MeV range. The highest energy plasma at the moment is supplied by a Hall accelerator. There is some possibility that this could be extended to $\sim$ 100 keV and hence might be useful if injection energies can be reduced.

Provision of plasma in an ignition experiment is likely to be difficult by any method other than ohmic heating. The crossed field injectors and button guns used on Proto-Cleo are very inefficient as the trapped fraction is small and only carries a few joules of energy. Hence plasma injection is likely to be a serious problem even with present experiments such as CLEO Stellarator.

THE APPLICABILITY OF INTENSE RELATIVISTIC ELECTRON BEAMS FOR IGNITION OF FUSION REACTORS

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Introduction

This paper considers the possibility of using intense relativistic electron beams (REB) for igniting toroidal reactors. Since they can act only as a very fast "match" they are not very useful for mirror reactors which require essentially CW heating. In spite of the rapidly advancing technology to produce REB's, we feel it is necessary to put into perspective the uncertainties both in the extrapolation of technology and of the physics involved.

A great deal of development has gone into REB machines (mainly for defence) and very large devices have now been built. The largest (known) is AURORA (built by / Physics International) which has four units, each giving a beam of 12 MeV, 0.4 MA for 100 ns, with four Marx generators storing a total of 5 MJ to produce ~ 2 MJ total beam energy [1]. The overall size is ≈ 18 m high, 40 m long. The vacuum coax guide that takes the energy of each unit to the beam producing diode has an outer conductor of 1.2 m diam. and a length of 4.5 m. Four diodes produce together the 2 MJ beam. Up to now there is no significant experience on the possibility to merge or unite several relativistic electron beams and to propagate the resulting beam over some distance.

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The system suggested here, as being necessary for igniting a reactor, would consist of 24 units each of 5 MJ beam energy (e.g. 25 MeV, 2 MA, 100 ns) with 24 × 10 MJ stored in Marx generators. Compared to the Aurora machine, the energy content of each unit is thus scaled up by a factor two roughly. J.C. Martin (AWRE), who pioneered such devices and has been concerned in their subsequent development, suggests that the overall size of each unit would be larger than Aurora by 30 to 50%. Thus, assuming all the technological problems of such an increase can be answered, the system would be very large indeed.

As far as the heating process is concerned, our arguments are based on the assumption that the return current will generate turbulence (as in normal turbulent heating experiments) leading to dissipation of most, or all, the beam forward energy. However, because there have been very few experiments on REB-plasma interaction, and none in toroidal devices, there is only limited (though apparently positive) experimental evidence of such heating [2]. In addition little is known about the behaviour at the head of the beam, which provides the driving force for the return current. Hence the method must definitely be placed in category C in Watson's list, i.e. "difficult to assess in the light of present knowledge". However, it must be emphasized that the extrapolation from present experience to the individual units considered here is relatively small compared with most other proposed heating schemes. Furthermore there is a fusion oriented development from high impedance machines like Aurora ($R \approx 30\ \Omega$) to generators with impedances of the order of $1\ \Omega$. These generators have the big advantage of less severe voltage problems and beam currents that are far in excess of the Alfven-Lawson limit.

Based on arguments presented in this note, the method of plasma heating by the injection of a REB is found to be applicable to low and medium β reactors with plasma densities around $n_p \approx 10^{20}\ m^{-3}$. High density reactors will not be considered here. High β reactors are excluded because the REB is preferably produced in a strong magnetic field.

Plasma and beam parameters

Fusion reactors [3] may have a plasma volume V of $10^3\ m^3$. With a plasma density $n_p = 10^{20}\ m^{-3}$ and a required additional plasma heating of $\Delta\kappa T = 2.5$ keV per particle, this demands the injection of U = 120 MJ of beam energy, assuming the beam energy is completely absorbed in the plasma. Supposing it is possible to build a diode delivering a beam with parameters $V_b = 24.5$ MV, $I_b = 2.1$ MA and $\tau = 10^{-7}$ s pulse length, we still need $N = 24$ injection points along the reactor, where N is found from energy conservation,

$$3\, n_p\, V\, \Delta(\kappa T_e) = U = N I_b\, V_b \tau. \tag{1}$$

REB parameters are conveniently expressed in dimensionless units. The beam particle energy is:

$$\gamma \equiv m/m_o = (1 - \beta^2)^{-\frac{1}{2}},\ \beta \equiv v/c,$$

where v is the particle velocity, m_o its rest mass and m its mass in the laboratory frame. The current is:

$$\nu/\gamma \equiv I_b/I_A,$$

where $I_A \equiv 17\ \beta\gamma$ kA is the "Alfven-Lawson limit". The beam parameters quoted previously correspond to $\gamma = 50$, $\beta = 1$, $\nu/\gamma = 2.5$. These numbers have to be understood as order of magnitude figures, that have to be optimized for any practical case. Other plasma parameters are assumed as given by Hancox [3]. For a toroidal reactor: major radius R = 12.5 m, minor plasma radius r = 2 m, toroidal magnetic field $B_T = 9.5$ T, peak magnetic field 14 T. The ions form a 50% mixture of D and T.

Beam requirements

Return current heating of a plasma imposes restrictions on the domain of possible beam parameters. In the following it is shown that the quoted beam parameters fulfil the requirements. According to a simple and idealized analysis [4] the beam particle density n_b, as observed in the lab frame, must satisfy the relation

$$(c_s/c) < (n_b/n_p) < (\gamma/2)^3\ (m_o/M), \tag{2}$$

where c_s is the ion acoustic speed, $c_s = \surd\,(T_e/M) = 0.62 \times 10^6\ \mathrm{ms}^{-1}$, for a final electron temperature of 10 keV; M is the average ion mass of the plasma. The left hand side follows from the requirement that the REB induced plasma electron drift velocity exceeds c_s. The right hand side of Eq. (2) corresponds to the requirement of a faster growth rate for the low frequency instability ($\omega \approx \omega_{pi}$) due to the interaction of drifting plasma electrons and ions than for the primary high frequency instability ($\omega \approx \omega_{pe}$) excited by the REB electrons interacting with the plasma electrons. Substituting numerical values into Eq. (2) we obtain

$$2.1 \times 10^{17} < n_b(\mathrm{m}^{-3}) < 3.4 \times 10^{20},$$
$$\text{or}\quad 10 < J_b(\mathrm{MA\ m}^{-2}) < 1.6 \times 10^4. \tag{3}$$

Eq. (2) illustrates the necessity of large γ values in reactor plasma heating. It also gives a minimum $\gamma = 4.4$ making $(c_s/c) = (\gamma/2)^3\ (m_o/M)$.

For beams with $\gamma > 3$, the space-charge limited diode current is given by

$$I_b(\mathrm{kA}) = 8.5\ (a/d)^2\gamma,\ \text{or}\ (\nu/\gamma) = 0.5\ (a/d)^2, \qquad (4)$$

where a is the beam or cathode radius and d is the cathode-anode spacing. For the beam under consideration, Eq. (4) gives $a/d = 2.2$. Supposing it is possible to build a diode with d = 11 cm, we find a = 26 cm and $J_b = 10\mathrm{MA\ m}^{-2}$. This current density is equal to the lower limit of (3) and the inequality is only marginally fulfilled. The associated final plasma drift velocity $v_D \approx c_s$.

Assuming that the low frequency ion acoustic instability saturates by the trapping [5] of D^+ ions the maximum electric field amplitude of the instability is given by

$$E_{max} = \frac{2M_1}{4e}\ \omega\ v_f\ , \qquad (5)$$

M_1 denotes the proton mass; ω and v_f are the frequency and phase velocity of the excited instability. Eq. (5) shows the importance of large plasma electron drift velocities as $E_{max} \propto c_s \approx v_D$. The numerical value of E_{max}, for the given beam and plasma conditions, is $\approx 10^5\ \mathrm{V\ cm}^{-1}$, giving rise to fluctuating potentials of a few k Volts. The development of the instability beyond the first saturation stage is completely open to further investigation. Assuming noise fields of the order of $1\ \mathrm{mV\ cm}^{-1}$, saturation is reached in about 1 ns. This time is short compared to beam pulse width.

The decay length L_n of the plasma current is given by[6,7]

$$L_n = (a/\lambda_E)^2\ v_b\ \tau_c,$$

where $1/\tau_c$ is a phenomenological collision frequency appropriate to the turbulent plasma, λ_E is the collisionless skindepth $\lambda_E \equiv c/\omega_{pe} = 5.4 \times 10^{-4}$ m. L_n gives the maximum length for the injected REB and the corresponding pulse length $\tau = L_n/v_b$. Taking $1/\tau_c \approx \omega_{pi}$ [5,8] we find as an order of magnitude value $L_n \approx 9 \times 10^3$ m and $\tau = 3 \times 10^{-5}$ s. With an actual pulse length of 10^{-7} s, these numbers indicate that the beam is current compensated during injection.

Eq. (2) is valid when there is both current compensation and charge neutralisation of the REB, i.e. when [7]

$$(a/\lambda_E)^2 >> (1 + \omega_{ce}^2/\omega_{pe}^2). \qquad (6)$$

where ω_{ce} is the electron cyclotron frequency. With $B_T = 9.5$ T, $\omega_{ce}/\omega_{pe} = 3$, so that Eq. (6) requires a $\gg 0.2 \times 10^{-2}$ m, which is well satisfied by our choice a = 26 cm.

Eq. (6) also defines the condition that a REB propagates in a straight line, unhindered by the ambient magnetic field [7]. Up to now there is no experimental support for this supposition. Instead it was found [9] that a fully space charge neutralized and current compensated REB, launched in a gas of 1 Torr, follows the magnetic field lines around a 180^o torus section. The beam only experienced a displacement caused by a $\nabla B \times B$ drift.

From Eq. (4) we observe that the beam current and therefore the beam density is proportional to γ. As the quoted beam only marginally fulfils Eq. (3) it is unlikely that return current heating of a reactor plasma is possible with a lower voltage beam. Beams with $\gamma < 50$ could still be used for heating, but these should rely on energy transfer with an only slightly enhanced collision frequency. The MHD stability of a REB in a torus becomes an important question in this case. The advantage of the method is that the REB contributes to the stability of the tokomak plasma.

Problems to be investigated

1. The feasibility and methods of injecting a REB into a closed magnetic field. This problem is most severe and its solution is strongly doubted by some of the authors. Although much work needs to be done only a few laboratories devote attention to this problem [9,10].
2. It has to be demonstrated that diodes can be constructed, with the small anode-cathode gap of d $\approx$ 10 cm, that work reliably in a magnetic field at voltages up to 25 MV. If the gap needs to be larger, REB heating of reactors might not work in view of Eq. (2) and (4), unless techniques are developed to focus beams in a magnetic field. J.C. Martin however, is doubtful about gaps with d < 30 cm for large γ beams.
3. Conversion efficiency of beam energy to plasma thermal energy and the partition of energy over ions and electrons.
4. The question whether the plasma heating is local or more uniform. With local heating there will exist regions with dangerously large temperature gradients.
5. Does the plasma become electrostatically charged by the injection of

an electron beam, or is the excess charge returned to the diode by the plasma return current?

6. Does the induced turbulence give rise to enhanced cross-field diffusion? Experience on turbulent heating experiments indicates that this is not a problem because the turbulent phase lasts much less than the Bohm time.

Only item 1 is specific to toroidal machines. All other points must be investigated for both linear and toroidal fusion reactors.

<u>General questionnaire</u> (See pull-out supplement at end; extract from Section B.1)

1. The required heating power is credible but involves considerable extrapolation. The coax feed that takes the voltage pulse from the Marx generator to the diode will be at the limits of presentday technology.
2. Electrical efficiency of REB generators is ≈ 50% [13].
3, 4. To be investigated.
5. No gas load problem.
6. Holes in vacuum wall will be large $\emptyset \geqslant 1$ m.
7. Although too early to answer properly, it might be necessary to place within the vacuum chamber both the diodes and/or windings to generate transient magnetic fields during REB injection. The hardware should be retractable.
8, 9. To be investigated.
10. Beam magnetic and electric fields will be mainly compensated for by the plasma.
11. To be investigated.
12. Present costs of REB installations, $10 per Joule of beam energy. May be reduced in future to ≈ 1$/J in case large numbers of installations are wanted.
13. Method is applicable to linear reactors and possibly to toroidal reactors when the plasma density is between 10^{20} m^{-3} and 10^{18} m^{-3}. Method will be used to start steady state reactors, as the rate of firing is limited by cleaning demands on the Marx generator, the feed and the diode.
14. The method of return current heating does not solve the reactor refuelling problem. Provided however that REB's could be absorbed in a suitably thin surface layer, DT pellets can be accelerated to velocities of order

10^5 ms^{-1} by present day REB's [11].

15. REB rings appear to have a long life time in a plasma. There is even a new reactor concept based on this property: the Astron-Spherator [12].

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The Reactor Refuelling Problem and the Possibility of a Pellet Injection Solution

Flemming Øster

The Refuelling Problem

If a toroidal reactor is going to operate in a steady-state it seems to be necessary to feed in fresh fuel close to the axis of the system to maintain the density profile. Several methods of refuelling have been proposed[1): charged particle injection, neutral injection, cluster injection, pellet injection, and others. The main difficulty with these processes is the high magnetic field and the hot, dense plasma across which the injection has to take place. Also the problem of helium exhaust[2) might favour other than injection solutions to the refuelling problem.

A solution to the problem that takes care of both difficulties mentioned is the gas blanket solution[3). In Ref. 3 it is proven that for an infinitely long cylinder a steady-state may exist where the fuel supply and the ash removal is undertaken through classical diffusion from and to the gas blanket. If the same picture holds for a toroidal system with more realistic transport coefficients this seems the most promising solution to the refuelling problem.

Here we shall be dealing with one of the injection methods proposed, namely that of solid, high velocity pellets. The advantage of this method to the other injection possibilities might be that a shield of gas and plasma develops by ablation. Thereby the remaining part of the pellet is protected from the fusion plasma, i.e. a lower injection energy per article is required.

Pellet Lifetime

The distance that the pellet has to travel in the environment of a fusion plasma before total disintegration is above 1 m. It is uncertain to which velocity a macroscopic pellet (radius

0.1-1 cm) can be accelerated, but one might take 10^4 m/sec as an upper limit. This velocity might be achieved with an induction acceleration tube[4] or through electrostatic acceleration. Additional acceleration by laser action on the pellets might be of importance also, especially for the case of very small pellets. Assuming the velocity mentioned the pellet lifetime, e.g. defined as the time in which half of the pellet is ablated, necessarily must exceed 100 μsec.

The simplest possible way to estimate the pellet lifetime is to assume that all the incident plasma energy flux is used in sublimation and ionization of the pellet particles. Sublimation cannot be the only process taking place; if so, a neutral layer of extreme density would immediately appear at the surface of the pellet. Assuming equal ion and electron temperatures in the reactor we take only the electron energy flux into account. Furthermore, we take constant density and temperature profiles. We obtain the ablation rate, G from

$$4\pi r_p^2 \frac{n v_e}{4} 2 kT_e = G (kT_{sub} + w_i)$$

where r_p is the pellet radius, kT_{sub} the sublimation energy, and w_i the energy used per ion pair created (w_i = 36.3 eV for hydrogen).

From the ablation rate we are able to estimate the lifetime of the pellet assuming no deceleration of the pellet to take place. Taking the reactor parameters as $n = 2 \cdot 10^{14}$ cm^{-3} and kT_e = 20 keV one obtains a lifetime roughly being a factor of 10 too low. In a work by Gralnick[5] the attenuation of the fusion energy flux due to heating of the ionized pellet particles has been taken into account. But even in this case the lifetime turns out to be at best marginal. Therefore, unless some additional shielding of the pellet takes place, this injection scheme does not look promising. Two different theories of additional shielding exist, an electrostatic and a magnetic one.

For equal ion and electron temperatures a pellet is charged up negatively under ion and electron bombardment if the secondary electron emission coefficient is small compared to unity. This effect was proposed by Spitzer et al.[6] and the increase in pellet lifetime was calculated to be roughly a factor of 10. It is seen that this electrostatic shielding mechanism depends on the secondary electron emission coefficient which is not known for hydrogen. But a comparison with other insulators leads to some optimism[7].

The magnetic shielding theory was developed by Rose[8]. The idea is that the cold plasma formed might create a β=1 boundary, thereby blowing a "balloon" in the main magnetic field. This effect will increase the pellet lifetime; it is however possible that the mechanism will be of importance only in the case of a high-β reactor[7]. The reason is that a solution to the equations given by Rose exists only if the β of the fusion plasma exceeds a minimum value. Taking the reactor parameters mentioned above the following minimum values for β are obtained.

r_p	1 mm	5 mm	1 cm
N	$2{\cdot}5.10^{20}$	3.10^{22}	$2{\cdot}5.10^{23}$
β	0.06	0.16	0.23

Experiments

Very few experimental results relevant to the present problem are available. However, some insight might be gained from a presently running study of the interaction between slow pellets and a rotating plasma[9]. The ion energy in the rotating plasma is about 1 keV and the density a few times 10^{14} cm^{-3}. The pellets have a linear dimension of 0.25 mm and after a short interaction phase the loss in mass is measured. The ablation rate found is very high compared with an estimate based on the simple sublimation and ionization model mentioned above. It is however not possible simply to draw the conclusion that the plasma energy does not have to be spent in ionization. This is due to the fact that the high radial electric field present, necessary to produce the rotation, might cause a high ablation rate as slow ions from charge exchange processes are accelerated and thus they cannot overtake the role of neutrals as for shielding purposes. Therefore, the interaction will be studied in further detail in the future.

The experimental results have also indicated a deflection of the pellets caused by the interaction. The electric field is maintained after the high energy rotational phase. Therefore, the deflection might be of electrostatic origin if the pellets are charged by the interaction. A further study of this is clearly of importance to one of the shielding possibilities mentioned above.

Conclusions

Theoretically the possibility of refuelling a steady-state reactor by injection of frozen pellets exists. Experimentally the results do not allow any firm conclusions to be drawn at present.

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THE GAS BLANKET CONCEPT

by

B. Brandt

A. THE OBJECTIVE

From general technological considerations fusion reactors with a closed magnetic configuration in a steady-state or quasi steady-state operation as stellarators, bootstrap-tokamaks or long pulse tokamaks have great advantages. However, fuelling and exhaust of these systems is difficult.

A blanket of high density around the thermonuclear plasma could possibly fuel the plasma by diffusion of hydrogen isotopes from the blanket into the thermonuclear core and if refreshed well enough could be sufficient to carry away the helium diffusing out of the core. As an additional benefit one can expect the blanket to separate the core effectively from the first wall preventing sputtering of the wall by energetic ions, charge-exchange neutrals and electrons. The blanket increases the heat flow from the core; however, this ionizes cold hydrogen and impurity neutrals on their way from the wall to the core, whereby direct charge-exchange and radiation loss from the core is decreased.

B. GENERAL DISCUSSION

Let us consider a cylindrical model. In the centre there is a thermonuclear core, burning D and T to He^4 and n. 80% of the gained energy is lost with the neutrons to the outer lithium shield. 20% of the energy carried by the He^4, has to maintain a steady-state temperature in which as much energy is carried away by the sum of all transport processes as is gained by the nuclear reactions and by other energy sources possibly present e.g. ohmic heating by the tokamak current.

Gas blanket[8,9] is the name of a zone surrounding the thermonuclear reacting core and extending radially out to the first reactor wall. The density is considered so high that the definition of a temperature and of a temperature profile are possible in the sense that the energy distribution of electrons, ions and neutrals at different radii can approximately be described by Maxwellians and at least in the outer most layers the temperatures of the different species are not too far apart. While certainly the radiation and probably other interactions like charge-exchange and ionization cannot be considered to be in thermodynamic equilibrium, so that Saha's formula and Boltzmann's niveau distribution law cannot be applied, it is required that enough interactions of repeated charge-exchange, ionization and neutralization take place to reduce considerably the number of particles deviating from the local Maxwellian distributions.

In the gas blanket the temperature decreases from the core to the wall; a decrease is also considered for the degree of ionization in the outer layers close to the wall[†].

Core and blanket are immersed in a strong magnetic field in the direction of the cylinder axis. The radial temperature gradient in combination with the axial magnetic field leads to a Nernst effect current vertical to both i.e. in the azimuthal direction. The magnetic field due to this (diamagnetic) Nernst effect current develops roughly in the classical diffusion time $\tau = \ell^2 \sigma_{\perp} B^2/p$ of a plasma in a field. It, therefore, has not yet been observed in tokamaks but there are strong indications for it

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† Whether a complete ionization in the radiation field of the thermonuclear core and the hotter inner layers of the gas blanket has to be expected and whether this would have a considerable unexpected influence on the gas blanket concept is being investigated. It certainly would have one advantage: the increased ionization would help to separate wall and hotter plasma regions.

in arc experiments. In a quasi stationary reactor the Nernst effect does develop. It leads to a radial gradient in the axial magnetic field. As $nkT + B^2/2\mu_o$ = constant, the plasma confinement is magnetic. The density still increases outward but much less than the temperature decrease would cause in a pressure contained isobaric plasma. One can get a central pressure around 50 atm in the reactor with an outer pressure in the Torr range. In the very outer layers with only partial ionization there is again a very marked pressure drop as neutrals diffuse inward easily but the outward ambipolar diffusion after their ionization is hindered by the B-field. In the temperature range from 20,000 K to 10,000 K this may be so pronounced that even the density drops.

C, REQUIREMENTS

Scientific requirements are:

1) There must be at least one combination of parameters which allow the core and the blanket to coexist. If too little nuclear power is produced the core will decrease and extinguish. At too much power the core will expand, burning away the desired blanket. There are corrective measures available from the outside - one can increase radiative losses by a higher He density or by deliberate impurity introduction and one can add energy by external heating.

2) It has to be proven that the above power balance is stable. Slightly subcritical operation of the reactor implying a relatively small external power source could bring down this problem to that of fast power control of the source only.

3) The normal requirements for stable equilibrium between plasma and external and internal (Nernst) electromagnetic fields have to be fulfilled (certain instabilities might be stabilized due to the presence of the blanket).

There are technological requirements e.g.:

The nuclear ignition of the core in the gas blanket is probably more difficult than ignition in absence of the blanket. Will it be possible at all? The gas blanket must keep its composition, can this be arranged?

Finally, there are economic requirements:

The cost of a reactor with gas blanket may not burden the delivered power unduly. This probably means that a unit length of reactor must not produce much less power that other systems now considered. The radius must not be much enlarged by the existence of the unproductive blanket.

D. THE VERBOOM/REM MODEL

In a time independent model of an infinite cylinder with a fictitious outside wall at a temperature of 10^5 K Verboom and Rem[1] calculated self-consistent solutions for a plasma fed by the α-particle energy of D-T reactions and losing the energy by heat conduction and Bremsstrahlung. The parameter space for these solutions is limited*. One of the first conclusions is the fact that the helium density on axis has to be kept higher than the density of D+T. Only then there is enough energy transport (by Bremsstrahlung) out of the thermonuclear core to prevent the core from overheating and expanding. The central temperature is about $5 \cdot 10^8$ K $\approx$ 40 keV; if the central pressure is 50 atm, corresponding densities are $n_e = 4.4 \cdot 10^{14}$; $n_D = n_T = 5 \cdot 10^{13}$; $n_{He} = 1.7 \cdot 10^{14}$ cm^{-3}. The power of the α-particles is 6 MW per meter reactor length; the total nuclear power then is 30 MW/m. The magnetic field on the axis is only 1 Tesla $\hat{=}$ 10 kG due to the Nernst effect, while at the outside a field of 3.8 Tesla has to be applied. As both the B-field and the kinetic pressure have a marked profile over the cylinder radius so has the usual *local* β. As here on the outside the kinetic pressure is negligible β could be defined as the kinetic pressure on axis divided by the

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* See Appendix I.

magnetic pressure at the wall. Then the model gives very high β's e.g. 0.86 for an axial pressure of 50 atm, and an axial field of 1 Tesla with a corresponding field of 3.8 Tesla at the outside.

It may be argued whether this β is acceptable for stability. As the plasma with the strong temperature and B-field gradient probably behaves quite differently from plasmas usually considered, one should not deny the possibility of this β right away.

If, however, one really needs a lower β, one can look at axial fields of 5 Tesla ($\beta \approx 0.35$) or even 10 Tesla ($\beta = 0.12$). Verboom and Rem have shown that as a consequence the radius of the thermonuclear reacting zone shrinks to 30% or even 15% of its original value with a corresponding decrease of the power per meter length of reactor. It may be that a compensating increase of R, the toroidal radius of the reactor, could still be accepted economically.

The numerical study is being extended into three directions.

a) An effort is made by Markvoort and Rem to include toroidal effects into the Verboom/Rem model. As a compromise between reality and computability the geometry of an infinitely long hollow cylinder as is used e.g. for the beltpinch is investigated. As a first result a diffusion similar to that in a torus with circular cross section was derived. For comparison the cylindrical calculations will be carried out with some arbitrarily chosen amplification factors in the diffusion coefficient simulating extra toroidal losses.

b) One of the limits of the Verboom/Rem model is the artificial wall temperature of 10 eV. Verboom[2] has also considered a cylindrical plasma column with a maximal temperature on the axis of 10 eV (Ohmic heating) and a wall temperature of 10^3 K. Goedheer will combine both models, looking mainly into the effect of the radiation from the core and the inner gas blanket on the outer layers with a temperature below 10 eV.

c) The time-dependence has to be studied.

E. EXPERIMENTS

Experiments [3-7] are of course far from thermonuclear models. We work on plasmas with densities and temperatures between tokamaks and arcs [7]. In Ringboog, a setup nearing operation, we hope to get a steady-state plasma with a fully developed Nernst effect at a density above 10^{15} cm^{-3} and a temperature about 15 eV, where we want to study the interaction between the central plasma and the outer layers. The upper density limit in tokamaks is an intriguing question in the gas blanket concept. We hope that with higher power machines and possibly with extra energy input in the outer layers, it will be pushed upwards sufficiently.

F. GAS PRESSURE CONFINEMENT

Budker, Kapitza, Alikhanov and others [10-15] consider thermonuclear plasmas confined not by the magnetic field but by the kinetic pressure of gas blankets in the pressure range of 30 atm and finally by the reactor walls. Magnetic fields remain necessary for the reduction of the heat conductivity. If these blankets were realistic they would be even more attractive that that considered here. For instance, in certain reactors even the neutrons could be absorbed in the gas.

However, the Nernst effect and the pressure gradient at the partly ionized boundary make the prerequisite at least very unlikely that a magnetic field could be found that decreases the heat conductivity sufficiently without interfering with the pressure balance.

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APPENDIX I (to Section B.7(b))

Self consistent solutions of a steady-state model reactor are found under the line $q_{P_o} = q_{R_o}$ where produced and radiated power are balanced on the reactor axis; a stronger limit is $Q_P = Q_R$ where their integral over the volume is balanced. In order to fulfill the requirement that the energetic α-particles have a gyration radius 4 times smaller than the half width of the temperature profile we have to work in the upper most bounded area in the graph. The α-particle power output of the model reactor then is 6×10^6 W/m (at the high temperature limit). The radial α-particle velocity is kept lower than the thermal velocity (lower right boundary), a sufficient criterion to prevent a singularity ($V_R = \infty$) warned for.

Other results were computed for 100 atm on axis and also for 5 and 10 Tesla. Moreover, there are profiles of temperature, density, magnetic field, and radial velocity and other information on the model.

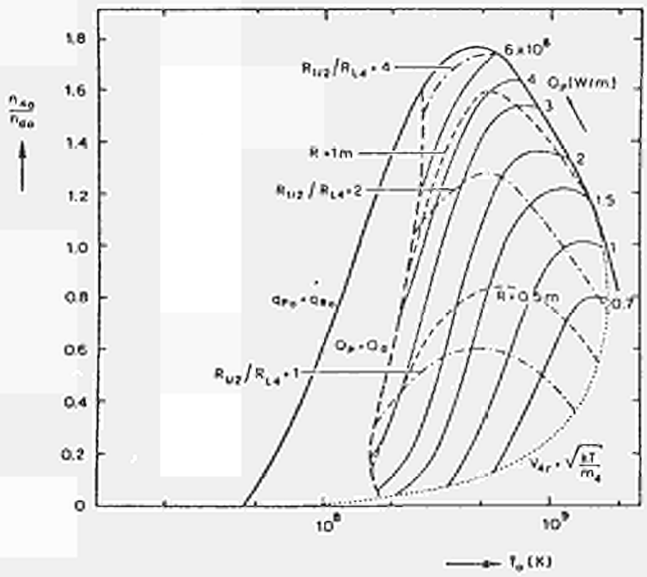

FIG. 2. Values of T_0 and $n_{\alpha 0}/n_{d_0}$ with $(d^2T/dr^2)_0 < 0$ for which solutions exist, $p_0 = 50$ atm and $B_0 = 1$ Tesla, with lines of constant power Q_P, constant radius R, and constant ratio $R_{\frac{1}{2}}/R_{L_\alpha}$.

(G.K. Verboom, J. Rem, Nucl. Fusion 13 (1973) p. 72)

Assessment Categories and Questionnaire (Extract from Section B.1)

Assessment Categories

A. Clearly in principle applicable: no foreseeable difficulties

B. Applicable provided that a few physical or technical difficulties can be overcome

C. Difficult to assess in the light of present knowledge

D. Inapplicable unless some unexpected breakthrough occurs

E. Clearly hopeless

Questionnaire

1. Is the required heating power available in this form, or at least credible?
2. How efficiently can power be created in this form?
3. How small can one make the losses in transit between source and plasma?
4. Are the consequences of the unavoidable losses acceptable (e.g. wall heating, degassing, sputtering)?
5. Is there an associated gas load problem?
6. What holes are required in the vacuum wall and are they compatible with neutronic requirements?
7. What hardware is required within the vacuum chamber; will it function in a reactor environment and what are its neutronic implications?
8. How does one ensure that the power permeates the whole plasma?
9. What fraction of the incident power is absorbed by the plasma and what happens to the remainder?
10. Do any electric or magnetic fields associated with the method affect particle containment directly?
11. What plasma distribution function results: is it stable, and if not do the instabilities affect plasma containment?
12. Can one estimate in order of magnitude the capital cost of all the hardware required per kW delivered to the plasma?
13. To which reactor concepts is the heating method applicable?
14. If none, is there nevertheless a case for using the method in the next generation of experiments?
15. Does the method also solve the reactor refuelling problem?
16. Does the method also maintain a plasma current which significantly improves the containment properties of the magnetic field?

C O R R I G E N D U M

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Alfred Nobel

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