CLM-P862

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Tokamak Advances Expected and needed and Advances in Exhaust Tech for Commercial Fusion Power in 21st Century

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AEA TECHNOLOGY

Contract with CEC: 361/98/1/FU UK NET

NET No. 89/197

The purpose of the contract was to:

(a) review the results for the maximum attainable beta, the confinement times and plasma currents needed, with special attention to JET and other large existing machines, and the advances expected and needed for reactor-sized machines;

make a comparison of steady state and pulsed tokamaks in view of fusion reactors;

(b) discuss the present state of divertor designs and the limits of performance;

consider the possibility of improving the performance by the use of higher heat load materials, or by reducing the heat load on the divertor plates by distributing the diverted power over the first wall or by radiating it by microwaves;

discuss the direct conversion of plasma energy into electrical energy.

I would like to acknowledge key contributions from:

P C Johnson
J Hugill
T N Todd
M Cox
A W Morris
and useful discussions with P Thomas

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Executive Summary on Tokamak Advances Expected and Needed and Advances in Exhaust Technology for Commercial Fusion Power Reactors in the Mid 21st Century

A review of the results of the maximum attainable β, current drive efficiency, confinement and divertor designs has been made. These have been related to a wide range of tokamak reactor studies which already exist and to potential reactors in the middle of the next century based upon possible advances in β, confinement, current drive and impurity control. These reactors range from conventional D-shaped plasmas to tight aspect ratio systems all of which lie just within the theoretical bounds of the first stability regime, except for one reactor design in the second regime of stability. This device is attractive as it only needs a current of ~ 5 MA but density limits and confinement then pose a problem as well as access to the regime.

The present reactor designs are considered to be rather too close to the present experimental and theoretical β limits for safe and reliable operation. An increase in accessible β is therefore desirable, apart from the clear economic advantages. There are several schemes to increase the maximum stable β at which a tokamak can operate. These are largely untested, but it seems likely that one or more will succeed, possibly to the extent of allowing aneutronic D-^{3}He designs. These configurations may deviate markedly from the base ITER/NET and reference design.

The assessment of confinement from present scaling is that there is a need for 2-3 times the conventional L mode scaling, which is compatible with the present improved confinement regimes observed on numerous machines either with X points or suitably tailored density profiles. It is thus so far concluded that the advances already demonstrated on devices such as ASDEX, DIII-D, TFTR and JET are adequate to make a range of tokamak reactors viable. Further advances in improving the confinement are foreseen, reducing the demands on current, current drive efficiency and size, though β and density limits will then become more important.

Steady-state operation of a tokamak reactor would reduce concerns over fatigue failure and might result in a cheaper design. Potential physics benefits include better stability, improved confinement and higher β limits. Recirculating power and economic constraints probably limit the acceptable driver power to $\simeq 10\%$ of the electrical output of the reactor which sets the minimum acceptable efficiency. Extrapolation to reactors of presently used schemes gives efficiencies $\eta \simeq 0.3$ - 0.6 which are just adequate for operation of a reactor with $I_p < 15$ MA albeit at densities which may be too low for operation of a high recycling divertor. Several schemes with higher theoretical efficiencies ($\eta > 1.0$) exist but waves with the required characteristics which can be used in reactor-grade plasmas have yet to be identified. In all cases ways of increasing the contribution of the bootstrap current would be desirable. An alternative approach using helicity injection either by modulation of the equilibrium or using waves offers very high efficiencies but remains to be demonstrated experimentally. Operation utilising cyclic variation of plasma parameters but maintaining constant plasma current would be possible even with relatively low current-drive efficiency. The benefits would then be smaller but a reactor cheaper than a purely inductive design might result.

For a range of power reactors with double null divertor configurations operating in the high recycling regime and with substantial Bremstrahlung and synchrotron radiation losses the target plate power density falls to acceptable levels (< 10 MW/m^2), ergodisation and divertor sweeping will help to reduce this further. However the high densities needed for the high recycling regime significantly reduce the current drive efficiency and synchrotron radiation loss. Both disruptions and ELMS will probably produce unacceptable erosion and their control is essential. There is significant potential to improve the power conversion efficiency with direct conversion of the diverted plasma and/or coupling the synchrotron radiation to MHD convertors.

Three particular approaches which would radically alter the conventional tokamak reactor concept were noted. These are the second stability regime at low plasma current with high aspect ratio which makes steady state operation using the bootstrap current likely, the tight aspect ratio approach with substantial β value and high power to mass ratio and the advanced conversion tokamak using synchrotron radiation to both drive the current and for direct energy conversion.

Tokamak programme advances in β, current drive efficiency, and confinement are summarised in Table 1.

Table 1

	present	expected	needed	future
Troyon coefficient, g (% m T/MA)	4	4	4	$\gtrsim$ 7-8
Current drive efficiency $\eta(10^{20}m^{-2}A/W)$	0.35	0.6	0.7	$\gtrsim 1$
Confinement improvement, τ/τ_G	2-3	3	2	4

1 Advances in β Limits Required for Commercial Power Reactors in the Mid 21st Century

1.1 Introduction

Reactor cost studies show that there is a substantial advantage from operating at high β. For example the ESECOM report notes that a reduction of 25% in β leads to a 7% increase in cost of electricity, whereas access to the second stability regime with $\beta = 20\%$ leads to a 16% reduction in the COE. Present operational β limits are in good agreement with optimised theoretical predictions with $\beta(\%) \leq gI(\mathrm{MA})/a(\mathrm{m})B_\phi^{vacuum}(\mathrm{T})$, with $g \simeq 3.5$ approximately independent of plasma shape. This is to be compared with the original work of Troyon, with $g \simeq 2.8$ and g =4–4.5 for ballooning optimised equilibria (no test of kink stability). β may thus be optimised by using shaped cross sections to maximise I for given a, B_ϕ and so high β can be obtained with high elongation, tight aspect ratio and low q_ψ.

A number of theoretical predictions have not been experimentally explored: in particular the second stability regime has not yet been convincingly attained. Current profile optimisation is so far doubtful and the limits appearing at high elongation have yet to be explored. In the near future all of these questions are expected to be investigated.

There are a number of encouraging possibilities for the long term, in particular the very high potential values of β in tight aspect ratio tokamaks; techniques for stabilisation of pressure-driven MHD instabilities as well as the possible attainment of second stability regime configurations.

1.2 Operational β Limits

1.2.1 Comparison of Experiment with Theory

The simple formulation of the first stability boundary hides a number of detailed but very important stability boundaries associated with rational values of q_ψ which lead to values of β significantly below the above limit, in particular with increasing elongation or, less importantly, with decreasing aspect ratio. Furthermore, some additionally heated tokamaks already experience β limits well below the Troyon limit, eg TFTR at high q_ψ ("supershots"), apparently because the plasma pressure and current profiles are far from optimum (these discharges are indeed at the maximum theoretical β for the measured profiles). Finally, the instabilities that appear at high β are often significantly different from the ideal-MHD ballooning and kink modes that are used to define the theoretical limit. Thus to some extent the agreement between experiment and the relatively simple theory of Troyon and others is fortuitous. It is also important to note that many devices see disruptions or confinement degradation well below the maximum β attained on that device. This is presumably largely an operational problem, but shows that a learning process at least is required.

1.2.2 Characteristics of Present Experiments

All the present experiments, in variously shaped plasmas exhibit operational limits with $g \lesssim 3.5$ but with anisotropic pressure or significant fast particle populations. There are deviants, TFTR at high q has a limit substantially lower and with a different scaling: $g_{max} \sim c - dq$ and is better represented by a limit on $\epsilon\beta_p$ or $\beta_p + l_i/2$ at high q. This is to be compared with hot electron plasmas on the TOSCA device at similar q values with $g \simeq 2.5$. Furthermore while the indentation of PBX and PBX-M allow very high values of β to be achieved for given current, the β-limit falls slightly below the value found on tokamaks with less extreme shaping ($g \sim 3$ as opposed to 3.5 achieved on DIII and DIII-D), and careful current programming (dI_p/dt large) is apparently required at low q_ψ to maintain favourable internal shaping. Strong internal shaping requires broad current profiles, which may be achievable in steady state with current drive.

The agreement with the theoretical limit suggests that at the β-limit either high-n ballooning modes, or else low-n ideal-MHD kink modes should appear. Experimentally the picture is not so simple: although some confinement degradation is sometimes seen when profiles are near the theoretical ballooning limit, there is so far little convincing evidence that high-n modes play any significant rôle (but there is dispute as to what "high-n" means). As far as the low-n kink modes are concerned, it is clear that in most cases the structure and growth rates imply that resistive or other dissipative terms must play a part.

Unfortunately at present there is very little good aspect ratio scaling data available (there is some from CLEO and TOSCA covering a factor of 2 in R/a but the β-limit was not well defined experimentally).

Operation at very low q_ψ ($1 < q_\psi < 2$) has been achieved and with strong additional heating but with some reduction in confinement through the q dependence.

The behaviour of the discharge at the β-limit also varies from machine to machine and with parameters for a given machine. For example at intermediate values of q_ψ the limit sometimes appears to be soft in PBX, ASDEX and T-10; the so-called "β-collapse", which may indeed be a useful feature in a reactor, whereas for $q_\psi \lesssim 3$ the limit is disruptive on all machines, usually with a resistive-MHD signature. Finally, at high q_ψ on TFTR the limit can be either disruptive (a very fast kink-like instability) or soft (strong internal MHD activity). Clearly it is desirable to avoid fast disruptions at high β in a reactor.

1.3 Comparison of Proposed Reactors with Theory

A selection of reactor studies are illustrated in Figure 1. This highlights the STARFIRE project as being substantially optimistic and even a number of the ESECOM cases are significantly above what we would consider safe on present experimental evidence. The reference reactor ("Reactor-2") is included, with $g = 4$ for the total β. It should be noted that there is the possibility that the α-particle β should be excluded, and may in fact exert a stabilising effect, thus allowing operation at larger g_{tot}. For Reactor-2 $g_{thermal} \simeq 3.0$.

Also shown is TPSS, a reference SSR reactor study, operating at low q_ψ with only 4MA of plasma current, $R/a = 6$, β=20% and $g = 22$. This value is apparently not the absolute theoretical maximum, the density (for reasonable T) and q limits probably being more important than the ideal-MHD stability. Some tight aspect-ratio schemes are plotted as well. Finally, a recent point from DIII-D is shown, demonstrating that β values in the reactor regime are already being achieved.

1.4 Relation to the Density Limit

Since $I/aB \sim 1/q_{cyl}$ for a given geometry, the β-limit can be represented by a third axis on the Hugill diagram. The β and density limits may be written:

$$\beta(\%) \leq g\frac{I}{aB} \qquad n_{20} \leq \alpha\frac{I}{\pi a^2 K}$$

where $\alpha \lesssim 1.3$. Combining these equations shows that the β limit is reached at subcritical density only for high temperatures ($\bar{T} > KaB \sim 20$keV) but there is significant sensitivity to the profiles. At lower temperatures the density limit is more restrictive and may limit the attainable β. It is known that changes in $n_e(r)$ can improve the density limit (peaking is good), but this may not be compatible with achieving large values of g.

1.5 Untested Predictions with Significant Influence on high-β Tokamaks

1.5.1 High Elongation

This helps in the simple Troyon formulation but seems on detailed examination to be bad due to destabilisation of both ballooning modes and kinks. A number of devices both existing and planned can test this.

1.5.2 Second Stability Regime

It is important to distinguish cases where the two regimes merge and there is a simple stable route to high β by heating alone from cases where careful transient programming of the current profile and shape are required to access a regime where the plasma is stable at high β but which is inaccessible by heating alone. The use of strong steady-state indentation or possibly small R/a fall into the first class, whereas the large aspect ratio proposals fall into the second. For the second class the most effective technique theoretically seems to be to raise $q(0)$, alternatives/additions being the application of an inboard separatrix or other favourable shaping. The SSR is usually discussed with reference to ballooning modes: it is far from clear whether there is a regime that is kink stable at arbitrarily high β, and the approach adopted by PBX-M for example is to use passive stabilisers to control the external kink mode.

There seems to be little evidence, either in sustained or transient tokamak plasmas or on fast pinches, apart from possibly TPE-2, that such regimes can be accessed. Resistive ballooning and tearing modes also seem to suggest possible difficulties in accessing such regimes, where extreme current profiles are required. However, if such a second stability regime were to become tenable then aneutronic D-^{3}He scenarios may be possible.

1.5.3 Current Profile Control

If operation at high g-values is required, then some control of the current profile may be essential. The feasibility is still questionable because it is hard to perform conclusive experiments. There are a number of results from existing facilities, eg the freeze and heat approach, and active control of the I_p which reveal the usefulness of control: a particularly striking example is PBX where a rising current is required to achieve high β and low q_ψ. It is not clear whether lower hybrid current drive or bootstrap profiles are good for optimising β but there is undoubtedly a need to be able to achieve high values of $q(0)$ for reliable control of the sawtooth.

1.5.4 Other Stabilising Techniques

There are a wide range of potential routes to stable operation at high β, eg fast particles, ponderomotive force etc. There are growing indications that such effects are real, but there is the danger of introducing new instabilities, such as the fishbone and its relations. There is also the possibility of using an active wall or some other magnetic feedback technique.

1.5.5 Intelligent Discharge Programming

This includes the use of localised heat deposition; careful control of the current rise phase; monitoring of derived parameters such the plasma internal inductance which has led to successful high current operation on JET; *etc.* Such tools would aid the reliable and safe access to high β.

1.5.6 "Infernal Modes"

These arise when the shear is low but the pressure gradient high. This can be the case in some ignition scenarios, and particularly second-stability tokamak schemes.

1.5.7 Resistive Modes

The complexity of the physics of these modes in high pressure, high temperature shaped toroidal plasmas has limited the study of these modes. As mentioned above, such modes apparently play a crucial rôle in the β-limit in PBX and DIII-D. There are indications that high temperature is beneficial.

1.5.8 Tight Aspect Ratio

There is a growing theoretical database at tight aspect ratio for proposed experiments, and this reveals that if the same definition of β is used, then the limit has a similar parametric form, the device remaining in the first stability regime with $\beta \gtrsim 20\%$. This then means that the β-limit may no longer be critical. It should be noted however that for such a device the magnetic energy can be predominantly in the *poloidal* field.

1.5.9 Low q Operation

It is important to operate at relatively low values of q_ψ, at least for conventional ITER-like designs. Increasing q_ψ from $\sim$ 2-3 to 4 would add $\sim$ 10% to the cost of electricity (ESECOM). Operation at still lower q_ψ allows higher β to be obtained for the same value of q, with concomittant reduction in cost. It seems from present experiments that the reliable disruption free attachment of $q_\psi \lesssim 2.1$ - 2.2, and in particular access to $1 \lesssim q_\psi \lesssim 2$ requires the presence of a good, close, conducting wall, or an actively controlled virtual wall. Confinement degradation may also be a problem, but this has not yet been explored sufficiently to assess whether it is a realistic option for a reactor.

1.6 Conclusions

Present reactor designs are considered to be rather too close to the present experimental and theoretical β-limits for safe and reliable operation. An increase in accessible β and g is therefore desirable, apart from the clear economic advantages. It seems likely that some significant improvement in present β values will be obtained in the near future with conventional designs. For example even while this report was being prepared β values of 8% have been obtained on DIII-D and 7% on PBX-M, but without increasing g. No doubt it will not be long before volume average values of 10% can be obtained in reactor relevant configurations in the tokamak. There are several schemes to further increase the

β and g at which a tokamak can operate. These are largely untested, but it seems likely that one or more will succeed, possibly to the extent of allowing aneutronic D-^{3}He designs. In particular recent theoretical work shows the potential for achieving high values of β ($\gtrsim 20\%$) at tight aspect ratios without active control or the presence of a conducting wall very close to the plasma. It is important that such approaches should be tested in the near term - some experiments are already under construction. In summary: there seems to be little doubt that there is significant scope for increasing the value of β in the tokamak configuration, appropriate to reactors in the middle of the next century.

2 Application of Non-inductive Current-Drive to Reactors

2.1 Introduction

The fatigue and reliability implications arising from the pulsed nature of inductive tokamak operation may detract from the viability of the tokamak reactor concept. Steady-state operation making use of non-inductive current-drive would remove these concerns but in turn increases the complexity of the device and may impose other constraints on the design or operation of the machine. The objectives of this section are three-fold: firstly, to set out the advantages and disadvantages of pulsed and non-inductively driven steady-state tokamaks; secondly, to review briefly the present experimental and theoretical status of non-inductive current-drive; and, finally, to identify the directions in which improvements to the present schemes might be found. Emphasis will be placed on generic issues rather than focussing on specific aspects of particular steady-state reactor designs or driver technology. It is important to note that the viability of non-inductive current-drive is strongly dependent on the plasma current necessary in a reactor to meet energy confinement and/or β requirements.

2.2 Comparison Between Inductive and Non-inductive Operation

The limited flux swing of the ohmic primary of a tokamak necessarily leads to pulsed operation. The length of the pulse naturally depends on the engineering design and plasma parameters but for conventional reactor designs ($R \sim 6m, R/a \sim 4$) lies in the range of $10^3 - 10^4$ seconds. Small major radius designs will tend to have relatively short pulses due to the smaller available flux swing. Inductive operation, therefore, implies a large number ($\geq 10^5$) of cycles during the 10^9s operational life of a reactor which makes fatigue failure a significant concern. Also, a means of maintaining the electrical output from the station during the transformer recharge periods may be necessary. Several modes of non-inductive operation have been proposed which might alleviate these and other problems associated with inductive operation. These can be broadly divided into two categories: those which are steady-state and those which utilise a cyclic variation of the plasma parameters to increase the overall current-drive efficiency.

The major engineering and economic advantage of very long pulse lengths made possible by non-inductive current-drive is that the reduced number of mechanical and thermal cycles during the machine's life almost eliminates fatigue failure which in turn allows higher power loading, longer component life or less demanding, and hence cheaper to meet, component specifications. The reactor economics are additionally improved by the elimination of the need for electrical supply levelling using expensive thermal and/or electrical storage systems and the reduced power supply and refrigeration requirements. It is also commonly assumed, perhaps without real substantiation, that the increased reliability expected from a steady-state tokamak may be decisive when marketing the tokamak reactor concept to the utilities. The main physics advantage offered by steady-state operation is that the radial profile of the plasma current may, in principle, be optimised by altering the parameters of the non-inductive drive scheme which then might offer improved stability, higher β limits (including easier access to the second region of stability), higher confinement and reduce any concerns over slow diffusion of the current profile in response to evolution of the plasma conductivity profile.

Inevitably there are also disadvantages associated with non-inductive operation. On the engineering and economic side, these include added complexity and increased need for component redundancy. Also, if the current-drive efficiency is low, the recirculating power fraction and power loading in the machine will be increased substantially which will lead, respectively, to a higher toroidal field requirement for a given electrical power output (at constant β) and more onerous demands on power exhaust components. Finally, more penetrations through the blanket may be required which would adversely affect the breeding capability of the blanket. The main physics disadvantage arising from non-inductive operation is that the need to optimise the efficiency of the current-drive scheme will require plasma conditions which are not optimised in other respects. For example, the efficiency of many current-drive schemes is improved by operating at low n_e and high T_e (see Section 4) which reduces the plasma reactivity at fixed β, may be incompatible with the use of a high recycling regime divertor, will increase the α-particle (and, if used, beam) β and may increase plasma transport by exacerbating trapped particle instabilities.

A cyclic scheme which offers improved overall current-drive efficiency by repetitively ramping the current in plasma conditions where the efficiency is high but the plasma is not ignited followed by a burn phase during which the current is allowed to decay inductively has been proposed (Fisch). Although this would allow indefinite sustainment of the plasma current, short cycle times ($\sim 10^2$s) are required if the current is to be kept nearly constant. The larger number of thermal and (albeit smaller) mechanical cycles compared with even inductive operation make this mode of operation seem unattractive. An alternative, hybrid scheme which uses a combination of an external transformer to maintain a constant plasma current during burn phases and current-drive under optimised conditions to recharge the ohmic primary, again at constant current, has also been suggested (Fisch). In principle, this allows a burn period longer than for inductive operation and maintains a constant current over many burn cycles. However, cyclic forces due to changing equilibrium fields and thermal cycling remain. A study by Ehst et al indicates that this scheme may be more attractive than purely inductive drive even for modest intrinsic current-drive efficiencies but is less attractive than steady-state operation (provided that the steady-state current-drive efficiency is high enough). An alternative hybrid scheme is to use partial non-inductive current-drive either during current ramp-up or sustainment to extend the length of the inductive pulse. However, the gains in pulse length involved are likely to be only a factor ~2 which may not be sufficient to justify the added complexity.

2.3 Limitations on Acceptable Current-drive Power

In order to set in context the efficiencies achievable using the various current-drive schemes discussed in the next section, it is necessary to establish the constraints on the maximum current-drive power. This will of course depend somewhat on the scheme chosen but five distinct types of constraint may be identified:

- economic limits to the recirculating power fraction. This is difficult to quantify exactly but it is likely that non-inductive drivers which increased the recirculating power by more than $\sim 10-20\%$ of the net electrical output would be unacceptable.
- machine power loading. In reactors the additional power to be exhausted is unlikely to be an insurmountable problem if the recirculating power is acceptable. However, there may be localised power handling problems associated with some schemes e.g. beam dumps in a neutral beam driven system.
- driver cost.
- loss of breeding blanket area due to penetrations for the drive scheme.
- possible reduction in useful β if current-drive schemes relying on high energy particles are used.

The precise limit set by these constraints will depend on the details of the reactor design and the current-drive scheme used. This has not been adequately considered in any design so far but the assumption generally made is that the driver power should not greatly exceed 10% of the electrical output of the reactor i.e. approximately 100MW for a typical reactor design. This implies that the minimum acceptable current drive efficiency, in amps per watt, is

$$\frac{I_{CD}}{P_{CD}} \gtrsim 10^{-2}\,\frac{fI_p(MA)}{wP_{elect}(GW)}$$

where f is the fraction of the plasma current supported by the non-inductive scheme (i.e. making allowance for the Bootstrap current) and w is the 'wall-plug' efficiency of the current-drive scheme. In conventional first stability regime designs $f \simeq 0.5 - 0.8$ although optimised profiles may give smaller f. Given the high β_p and lower currents typical of second stable regime devices the high bootstrap contribution may remove the need for current-drive. At present generation efficiencies of $w \simeq 0.7$ are expected. In terms of the normalised current-drive efficiency usually quoted this becomes

$$\eta_{CD}^{min} = \frac{\bar{n}_e R I_{CD}}{P_{CD}} \gtrsim 10^{-2} \frac{f \bar{n}_e (10^{20} m^{-3}) R(m) I_p(MA)}{w P_{elect}(GW)} = \frac{f \eta_{eng}}{w}$$

Values of $\eta_{eng}/\bar{n}_e$ for various reactor designs are set out in Table 1 showing the need to achieve $\eta_{CD}^{min} \gtrsim 0.7$ for typical values of f and w in first stable regime D-T reactors operating with $\bar{n}_e \simeq 10^{20} m^{-3}$. The ESECOM study highlights the need to achieve the highest possible efficiency; reducing η from a reference value of 2.7 to 0.5 increased the reactor cost by up to 40%. In a second stable device such as TPSS the large boot-strap contribution and small current required lead to $\eta_{CD}^{min} \simeq 0.06$. The high current of the $D -^3 He$ device results in $\eta_{CD}^{min} \gtrsim 5$.

Table 2

	STARFIRE	EEC REFERENCE 2	ESECOM V-Li	ESECOM V-MHD	ESECOM V-$D^3 H_e$	CSTR	TPSS
I(MA)	10	16.6	16	25	60	20	4
R(m)	7.0	5.3	5.9	5.0	8.6	1.6	6.0
$P_{electrical}$ (GW)	1.2	1.2	1.2	1.2	1.2	0.9	0.9
$\eta_{eng}/\bar{n}_e$ ($10^{20} Am\ W^{-1}$)	0.58	1.07	0.79	0.64	4.3	0.36	0.27

2.4 Current Drive Efficiencies

A large number of non-inductive current-drive schemes have been analysed which use either particle injection or resonant wave-particle interactions (see, for example, the review by Fisch). In addition, methods which rely on the non-linearity of the Lorentz force (commonly referred to as helicity injection schemes), originally identified as non-inductive current-drive schemes for RFPs, have been proposed for tokamak applications. A brief review of the most promising schemes for reactors will be given here in two parts. Firstly, the efficiency of the underlying physical processes and its scaling with parameters will be discussed; this defines the fundamental limits to the efficiency achievable using the scheme independent of any technical considerations. Secondly, the factors limiting the maximum efficiency achievable in practice will be outlined. The value of dividing the task into these two aspects is that the first part reveals the ultimate potential of the scheme and the second identifies the key factors which are preventing the full theoretical efficiency being achieved.

2.4.1 Inductive Current Drive

The efficiency of inductive current-drive sets a convenient standard against which other schemes may be judged. At reactor temperatures the plasma resistance ($\propto T_e^{-3/2}$) is very low and consequently the efficiency $I/P = 1/V_{loop}$ is very high - typically $\sim 30AW^{-1}$ for thermal plasmas; this would be increased if a significant non-thermal population was present. The highest efficiency achieved so far in thermal plasmas is $\sim 3AW^{-1}$. Furthermore the efficiency is almost independent of n_e unlike the resonant wave and particle injection schemes discussed below whose efficiencies scale as n_e^{-1}. Application of inductive current drive to reactors is straightforward except that to maximise the pulse length, the field in the ohmic primary will be pushed to the limit which may result in the coil having to be replaced several times during the life of the reactor. Also, large major radius reactors are favoured since the increase in area available for the ohmic primary more than compensates for the increase in loop voltage for given plasma current. Small aspect ratio, inductively driven reactors are not favoured, due to their higher resistivity arising from neoclassical effects, unless they approach omnigeneity.

2.4.2 Helicity Injection

Four distinct types of scheme have been proposed which utilise the non-linear nature of the Lorentz $\underline{v} \times \underline{B}$ force to generate an effective electric field in the plasma. Efficiencies approaching the ohmic value are predicted in some circumstances.

Equilibrium Schemes

Modulation of the equilibrium using a combination of oscillating loop voltage and toroidal flux (achieved by altering plasma elongation or radial position) can, in principle, generate currents in tokamak plasmas if relaxation processes occur. Although the power dissipated in the plasma would be small, since the efficiency approaches the ohmic value, there will be a large reactive component seen by the power supplies which leads to large power supply requirements and the risk that eddy current losses will dominate the energetics. Eddy current heating of superconducting coils may also be problematic. Attempts to drive currents on DIII-D have shown increased rate of decay of current when the phasing was set to drive currents in the opposite direction to the plasma current but little positive current drive apparently due to impurity production. An alternative approach is to inject helicity using electrodes as in CDX where a tokamak discharge has been sustained. Experiments on larger tokamaks are being planned.

Wave Schemes

Schemes that utilise the non-linear nature of the Lorentz force have been investigated for a wide range of wave frequencies (Bhadra and Chu; Ohkawa and Chan; Assis and Lashmore-Davies). At low $\omega \sim \omega_{ci}$ it is necessary for the waves to carry helicity (i.e. to be circularly polarised) which may be difficult to arrange. At high frequency wave helicity is not required. At all frequencies it is necessary for the wave to be damped for a current to be driven. The current is carried by the electron fluid but the wave damping required can be provided by either the electrons or ions; resistive damping by electrons by itself will, however, probably be insufficient to drive significant current. Resonant wave damping may be utilised but the overall efficiency of the scheme has not been considered in detail yet. Initial theoretical indications are that efficiencies approaching ohmic values may be obtained at low frequencies ($\omega \sim \omega_{ci}$) although some schemes retain a density dependence. At high frequency ($\omega > \omega_{ce}$) the efficiency is lower than ohmic by a factor $\sim v_d/c \sim 10^{-3}$, where v_d is the ohmic drift velocity, except near wave cut-offs and at the cyclotron resonance of the whistler wave where the efficiency becomes large. No experiments have been performed yet so the practical efficiency is unknown.

Fluid Entrainment

An alternative approach (originally used by Thonemann et al) which relies on the electron fluid being entrained by a rotating magnetic field offers essentially ohmic efficiency. Several devices have demonstrated current-drive eg Rotamak, Synchromak. Experiments on the small Rythmac device show that a tokamak configuration can be sustained but no experiments have yet been performed on high temperature tokamaks.

Injection of Compact Tori

It has been suggested that high β compact tori could be injected into tokamak equilibria and thereby allow steady-state operation by supplying helicity to the discharge. This scheme also offers a means of refuelling the plasma.

2.4.3 Resonant Electron Schemes

Within the framework of non-relativistic collision theory and neglecting trapped particle effects, wave schemes which cause electron diffusion in parallel velocity (Landau damping and transit time magnetic pumping) give current drive efficiencies which asymptotically scale as $v_{\|res}^{-1} T_e^{3/2}/n_e$ and $v_{\|res}^2/n_e$ at, respectively, small and large values of $v_{\|res}$. At low velocities the efficiency approaches the ohmic value when $v_{\|res}$ is comparable to the ohmic drift velocity. (Here $v_{\|res}$ is the velocity component of the resonant electrons along the magnetic field.) TTMP is somewhat more efficient than LD since the waves interact with higher $v_\perp$, and therefore less collisional, electrons. Cyclotron damping of waves leads to diffusion in $v_\perp$ which drives current via the Fisch-Boozer asymmetric collisionality process. At large $v_{\|res}$ the efficiency for $\omega \sim n\omega_{ce}$ is 0.75 times that of LD schemes but at low $v_{\|res}$ its efficiency is small. Strongly doppler shifted cyclotron damping of low frequency ($\omega \ll \omega_{ce}$), high parallel momentum content (e.g. whistler) waves on fast electrons would give very high efficiencies. When particle trapping is included the efficiency of LD and TTMP schemes falls dramatically at low $v_{\|res}$ because wave interaction with a trapped electron can increase its canonical angular momentum but cannot drive a toroidal current directly; the efficiency with which canonical momentum can be converted to toroidal current appears to be too small to be of interest.

For all resonant electron schemes, therefore, interaction with fast electrons is favoured which at reactor relevant temperatures makes relativistic effects important. For LD the efficiency improves with $p_{\|res}$ but less rapidly than in the non-relativistic theory. For cyclotron damping the efficiency reaches a maximum for $p_{\|res} \simeq m_e c$. In both cases the efficiency increases weakly with temperature for $T_e < 50keV$. For a $25keV$ plasma the efficiency for LD reaches $\eta \gtrsim 2$ at velocities $\gtrsim 0.97c$ while the peak efficiency for cyclotron waves is $\eta \simeq 0.65$. The overall efficiency in each case will be somewhat smaller than these values due to the need to sustain an extended tail on the electron distribution.

The lower hybrid wave scheme, which relies on LD, has demonstrated current drive efficiencies consistent with theory in many machines. Recent results from JT-60 have demonstrated $\eta \simeq 0.35$ which is as predicted for the $\sim 200keV$ electron tail energies involved. The launched spectrum used has the small spread and good directivity needed for high current drive efficiency and the fastest (most efficient) waves consistent with accessibility to the plasma have been launched. The scaling of efficiency with wave parameters is consistent with theory which suggests that upshift of the wave spectrum is not a significant issue in JT-60. Little improvement over the present result is therefore possible. Present theoretical work indicates that the beneficial effect of higher temperatures in reactor plasmas is counter-balanced by their poorer wave accessibility and overly strong wave damping which leads to current-drive only towards the plasma edge. Damping on α-particles and beam ions is also a concern. High intensity pulsed power or narrow spectrum waves may enhance the penetration of the waves but the accessibility constraint remains. Local generation of waves via the beat-wave process reduces the accessibility problems but the overall system efficiency is not significantly improved. A range of efficiencies from $\eta = 0.3 - 0.6$ covers all of the theoretical predictions. These values fall well below the $\eta \simeq 2$ limit imposed by relativistic dynamics and therefore there is considerable scope for improvement. Ways of launching waves which have good accessibility to reactor grade plasmas at phase velocities close to the speed of light and with good damping need to be identified. The fast wave at low frequency ($\omega \simeq \omega_{ci}$) offers good accessibility at high phase velocity but its damping (via TTMP) is predicted to be weak even in high T_e, high β plasmas. In order to obtain sufficient damping waves slower than those needed for optimum current-drive would have to be used with the consequence that the efficiency falls to $\eta \simeq 0.3 - 0.6$. Conclusive experimental results are not yet available.

Present results from electron cyclotron current-drive experiments are inconclusive. Thermal plasmas heated at $\omega = 2\omega_c$ on CLEO showed efficiencies a factor ~ 3 lower than theory. (In the W VII-AS stellarator, however, the observed current-drive efficiency appears to be close to the theoretical value.) Results from WT-2 and WT-3 at both ω_c and 2 ω_c using non-thermal plasmas showed very low current-drive efficiency. For reactor conditions, upshifted (outside launch) schemes are favoured over down-shifted schemes to minimise wave enhanced trapping effects (due to the relativistic resonance condition). Efficiencies predicted for this scheme are $\eta \simeq 0.2 - 0.3$. Experiments are needed in conditions where the wave damping is very strong (as it will be in reactors) to see whether the low efficiency in present experiments is simply due to the insufficiently strong damping. In high field, high T_e reactors the synchrotron emission may be sufficiently intense to drive a substantial current if the walls of the device have toroidally asymmetric reflectivity. Single-pass non-quasi-linear acceleration or stochastic heating of electrons to very high energies using high intensity radiation from FELs may allow the efficiency of electron cyclotron current-drive to be increased to $\eta \sim 0.6$.

2.4.4 Relativistic Electron Beam Drive

There are two ways of expoiting injected relativistic electron beams (REBs) for current-drive. Firstly, if the plasma resistivity is assumed to be unchanged by the beam injection, the beam loses energy mainly by collisions and the efficiency is similar to that obtained with wave schemes which cause parallel diffusion of electrons at the same energy as the beam. This offers an alternative means of reaching the the high efficiencies ($\eta \simeq 2$) mentioned above. Alternatively, if an intense, pulsed REB capable of driving (if it were steady-state) a current much greater than the plasma current is used and the plasma resistivity is greatly enhanced during the pulse (e.g. due to bump-on-tail instabilities), the energy is transferred almost entirely to the poloidal field. If the plasma resistivity then returns to the (neo)classical value, the decay of the current increase generated occurs on a slow L/R time and the time-averaged efficiency is the same as for inductive current-drive. Methods for transporting the REB into the plasma are needed.

2.4.5 Resonant Ion Schemes

The efficiency of generation of current using ion schemes is smaller than with electrons by a factor $\sim (m_e/m_i)^{1/2}$ so there is little prospect of their use for bulk current drive in reactors.

2.4.6 Ion Beam Drive

The physics of ion beam schemes is well understood and experimental results agree quantitatively with theory. The efficiency of ion current generation using D injection into a D-T plasma peaks at an energy $\sim 90T_e$ but is within $\sim 20\%$ of the peak value for energies within a factor 2 of the peak. The efficiency scales as T_e/n_e i.e. $\eta \propto T_e$ (in contrast to electron schemes) which strongly favours low n_e, high T_e operation; for $T_e \simeq 20keV$ the efficiency of <u>ion</u> current generation for parallel injection is $\eta^i \simeq 0.9$. The reverse electron current arising from electron-ion collisions reduces the nett efficiency of the scheme to $\eta \simeq 0.7$ for typical Z_{eff} and ϵ although the reduction on axis $(= 1 - 1/Z_{eff})$ is somewhat larger. Injection of a beam as near parallel to B (to maximise the parallel ion current) and ionisation inboard of the major radius (to minimise trapped ion effects) are favoured which requires good beam penetration. For conventional reactor designs operating at $n_e \simeq 10^{20}m^{-3}$ this corresponds to an energy of $\sim 1 - 2MeV$ which is fortunately close to the optimum for current drive. In typical applications the inability to inject parallel to the field and trapped particle effects reduce the overall efficiency achieved to $\eta \simeq 0.3 - 0.5$ for $T_e \simeq 20keV$. Use of high Z beam ions would give a slightly higher efficiency. The value $\eta = 0.7$ assumed for the EEC 'Reference 2' reactor appears to require a modest improvement beyond present predictions.

2.5 Conclusions

All of the resonant wave and beam schemes presently used would give $\eta_{CD} \simeq 0.3 - 0.6$ when extrapolated to a reactor (assuming $T_e = 10-30keV$, $n \sim 0.5-1.5\times10^{20}m^{-3}$). This is <u>just</u> adequate for a 15MA reactor (e.g. $n_e = 0.7 \times 10^{20}m^{-3}$, $R = 6m$, requires $P_{CD} \simeq 100 - 200MW$) if the bootstrap current is ignored. In the first stability regime, inclusion of the bootstrap current would reduce the power required by $\sim 30 - 50\%$. Optimisation might allow even larger contributions from the bootstrap current. Ways of increasing the efficiency of resonant electron schemes towards their theoretical limits ($\eta \simeq 2$ for LD at $T_e = 25keV$, $v_{\|res} = 0.97c$) would be desirable. However, ways of generating waves with appropriate accessibility and damping characteristics are lacking at present. High parallel momentum waves which cyclotron damp on fast electrons would also give high efficiency but a means of launching the waves needs to be identified. Other schemes also offer high efficiencies. Relativistic electron beams can approach ohmic efficiency but cross-field transport of the relativistic electrons into large tokamaks needs to be demonstrated. Low frequency helicity schemes are also predicted to have efficiencies close to ohmic values. High frequency helicity schemes are not significantly more efficient that resonant schemes except near resonances and cut-offs. All helicity schemes require experimental verification. Cyclic current-drive schemes would be possible for even quite low efficiency current-drive but the operational benefits are smaller. If operation in the second stability region is possible, the low plasma current required and the large contribution from the boot-strap current would make current-drive very easy.

Given the difficulty of achieving sufficiently efficient current-drive in a reactor it is really necessary to resolve pulsed versus steady-state reactor arguments. Ideally a single, impartial design team should compare reactor costs etc of the two approaches. This will of course depend on the current-drive efficiencies achievable. Areas which merit further investigation include:

- Finding feasible wave schemes which can drive very fast electrons in a reactor (i.e. are accessible and have reasonable damping). Waves which have large parallel momentum and interact with fast electrons deserve special attention.
- A detailed study of helicity injection schemes (including wave injection).
- Investigation of means of maximising the contribution of the boot-strap current.
- Reactor designs aimed at optimising current-drive efficiency e.g. having the lowest I_p and R consistent with τ_E and β scalings. Small aspect ratio reactors which approached omnigeneity would be worth consideration because the trapped particle effects would be smaller.
- Full system efficiency calculations i.e. what is the best overall efficiency achievable (including likely generation efficiency, directivity, trapping).

3 Confinement Issues for Tokamak Reactors

3.1 Importance of Confinement

The size/cost of a fusion reactor will be set by the parameters required to achieve the necessary confinement, density or beta performance. Which of these three constraints dominates depends on the scalings applicable in a reactor. Present indications are that the Troyon β limit ($\beta_T(\%) = 0.035$ I(MA)/a(m)B(T)) and the Hugill-Murakami density limit (n(10^{20}m^{-3}) = 1.26j (MA m^{-2})) can be reliably used to predict the operation limits in a reactor. There are, however, many confinement scalings for different plasma regimes, leading to different requirements for reactor parameters, in particular the current. It is not obvious which regimes are relevant and optimal.

3.2 Effect of Confinement Scaling on Ignition Margin and Plasma Current

To illustrate the effects of different confinement scalings we will take Goldston L-mode scaling and neo-Alcator scaling to represent the range of possiblities. Note that using a simplified O-D model which neglects profile and dilution effects; if the thermonuclear power ouput is approximated by

$$p_{TN}(\mathrm{MW/m^3}) = 7.9\mathrm{x}10^{-3} n(10^{20}\mathrm{m}^{-3})^2 T(\mathrm{keV})^2, \tag{1}$$

then the required confinement time is

$$\tau_E(\mathrm{s}) = \frac{30.5}{nT}. \tag{2}$$

L-mode Scaling

For L-mode confinement the ignition margin scales as $nT\tau \propto I^2 A^{2.5} a^{-.24}$

For A=3, and a $\simeq$1 the plasma current needs to be $\simeq$ 36MA for $\tau_{E-G} = \tau_E$. However, a factor of 2 improvement in τ_{E-G} by appealing to H-mode confinement reduces I_p by a factor of 2.

Neo-Alcator scaling

For neo-Alcator scaling, the field strength and current do not enter explicitly and there is no confinement penalty for increasing n or T, so the additional constraints imposed by β and density limits must be applied. The ratio of neo-Alcator to the required confinement then becomes

$$\frac{\tau_{N-A}}{\tau_E} = 5.72 \times 10^{-5} \frac{I^3 A^3 q^2}{K^2 a} \tag{3}$$

Taking A = 3, a $\simeq$ 1, q $\simeq$ 2, gives a required plasma current of $\simeq$ 5.4 MA in a circular plasma or 8.6 MA with K = 2 if the density and beta limits are reached simultaneously. Thus 5 MA represents the minimum current (at $A = 3$) for which a tokamak reactor could work for our present limits on density and β and best confinement. A confinement scaling somewhere between the two eg $\tau \propto n^{\alpha} I^{\beta}$ but with power degradation ($P^{-\gamma}$) seems appropriate at the present time. This covers strong ECR heating experiments where $\alpha \sim 1$ and beam heating and strong ICRH experiments where $\beta \sim 1$ except supershots where $\beta \sim 0$. γ can be less than 1/2 in H mode (DIII-D) or supershot (TFTR) conditions. This indicates that low currents may be feasible.

In all these scalings the dependence on aspect ratio is crucial, but unfortunately the confinement data base is poor. The neo Alcator expression seems to be well borne out in medium scale facilities for A $\sim 3 \rightarrow 7$. The aspect ratio dependence in the Merezhkin scaling has been obtained from strong additional heating (NBI and ECRH) from devices with a two-fold range of aspect ratios. However, apart from the accidental variation in different experiments which, due to technical improvments in machine design, tends to correlate reducing aspect ratio with larger, more modern devices with better diagnostics and heating methods, the only controlled experiments in the same device tend to be flawed by changes in plasma-wall spacing and impurity content.

There are three aspects associated with strong α-heating: first the confinement of the energy of the fast particles, second the possibility of anomalous loss of α's and finally the effectiveness of electron heating.

ICRH and high energy NBI experiments create significant fast particle populations, and ICRH minority species can simulate α-particles. There is no evidence for anomalous loss of fast particles if the fishbone regime is avoided. The fast particles pressure does not appear to degrade confinement. Experiments with strong ECRH at low density with large banana-orbits do not show a degradation of confinement with power, and strong electron heating may even be beneficial. Large populations of super-Alfvenic α-particles have yet to be studied of course.

Profile peaking effects are also very important in obtaining global ignition as core ignition can be obtained by satisfying eq (2) with the central values of n, T and τ_E, thus reducing the demands on global confinement considerably if a stably propagating burn can be obtained. A global Q of 1 is sufficient for core ignition given presently observed peaking factors and values of $\tau_E/\tau_E(O)$. Such profiles combined with present H mode confinement scalings would reduce the required plasma current to less than 10 MA.

3.3 Prospects for Confinement Improvement

In the above discussion we have tacitly assumed that confinement scalings observed in smaller devices can be extrapolated to the reactor regime. The justification for this assumption must be examined. It is easily shown that no experiment can be built with the same Connor-Taylor parameters as a reactor, which has less power output and is cheaper. Thus there is no justification for the assumption that the physics of a reactor-scale plasma can be reproduced simultaneously in any one experiment. However, tokamaks have been operated at all relevant values of all the appropriate dimensionless quantities. There are no signs that any of them, either singly or together, represent a threat when extended to the reactor regime. Furthermore, the remarkable success of the L-mode scaling established on small devices (up to the size of PLT) in fitting results from the next generation, such as JET, gives some confidence that it, and other operational scaling can be extended further.

It is equally clear that this is not the whole story. For a given set of externally imposed global parameters a variety of methods has been found to increase the energy confinement given by L-mode scaling by up to a factor of 3. Perhaps the best established of these, giving typically a factor of two increase, is the H-mode, usually obtained using a poloidal field separatrix and additional heating, though neither of these is strictly necessary. The confinement improvement obtained in the H-mode seems to be associated with a thermal barrier in the plasma periphery though this is less obvious in limiter H modes. The scaling of the confinement is similar to that in the L-mode, at least in H-modes which are free of ELMs. The favourable isotopic mass dependence should also not be forgotten for D-T plasmas.

The second way in which confinement has been improved is by using a variety of techniques which result in a more peaked density profile. These include pellet injection first on Alcator on a range of other devices and extended to high temperatures ($\sim$ 10 keV on JET) reduction of edge recycling and/or gas feed (B-mode on T-10, IOC on ASDEX) counter-injection of neutral beams (some indications from ORMAK onwards; clear results from ASDEX) and central refuelling together with reduced edge recycling by balanced injection (TFTR supershots). These techniques usually improve the confinement at higher density, for example, in the so-called saturated regime of ohmic confinement, where a large fraction of the power loss is via the ion channel but JET and TFTR have shown that effects can be produced at high temperature. It has been postulated, with some direct supporting evidence from, for example, the TEXT tokamak, that the confinement improvement results from the stabilisation of the ion temperature-gradient mode.

An obvious requirement for reactor scale devices, in maintaining a peaked density profile, is to obtain sufficiently good penetration by pellets, particle beams or perhaps injected spheromaks to refuel the plasma centre. It is unclear at present whether the benefits of peaked density profiles and H-modes can be combined though the supershots certainly show that improved confinement and peaked profiles are compatible. Indeed, it seems that the scaling of τ_E with plasma parameters is not the same in the two cases as if different processes are involved. In the improved ohmically heated regimes and in plasmas where ion heating is negligible, such as those in T-10 obtained with powerful ECRH, a strong dependence of τ_E on density is observed and a much weaker dependence on current, the reverse of the case in the H-mode. The reasons for these differences need to be investigated further.

A third technique is to control the MHD activity in particular the internal sawtooth activity which can be effected by a variety of scenarios using localised RF heating and leads to improved core confinement.

The apparent strong aspect ratio scaling suggests a possible link with trapped particle effects which can be influenced to some extent by magnetic geometry, shaping and high β effects suggesting possible means of reducing their impact in the future (eg Ohkawas comment). The dependence on plasma shaping particularly at high elongation ($k \sim 2.5$), as in many reactor designs is at present unknown.

Those reactor designs that feature a superthermal electron population as part of their current drive scheme may benefit from improved confinement as seen as slideaway discharge on CLEO and ASDEX amongst others.

3.4 Particle and Impurity Confinement

Measures employed to improve energy confinement may also result in improved particle confinement, which is not always wanted, and impurity confinement, which is never wanted. Some method of independently controlling particle and impurity transport and their sources and sinks will be necessary. For example the tendency of impurities to accumulate in well-confined discharges is mitigated in present experiments by instabilities such as sawteeth and ELMs (eg recent 5s H-modes on DIII-D). Improvement in energy confinement or peaking of the pressure profile by stabilising such instabilities may have to address this problem of impurity control though there was no evidence for impurity accummulation over 5 sec in PLT with sawtooth stabilisation by LHCD. Ash removal may also be aggravated by these measures although exhaust of slower alphas could be facilitated by pumping them into ripple loss orbits by direct ICRH.

Our present understanding of the relationship between energy and particle transport coefficients is insufficient to enable effective strategies for independent particle control to be proposed. However, there is some indication from experimental results that such techniques will be found. For example there are differences in impurity transport in some conditions produced by co- and counter beam injection, as compared with their effects on energy confinement. Advances in divertor and limiter technology and materials (eg island divertors or separatrices, ergodic boundary, doped composite graphites etc) will play an important role. There is ample evidence that divertors can affect particle sources and sinks.

3.5 Conclusions

There are considerable prospects of improving the confinement time in the short term by a factor of 2-3 thereby reducing the necessary plasma current to $\lesssim 15$ MA. However β and possibly density limits then pose a threat of disruptions necessitating some form of active control. Further improvements in the longer term by a variety of techniques towards optimised ohmic confinement times can be foreseen which would allow significantly lower currents ($\lesssim 10$ MA) and even currents as low as 5 MA which are just within the limits of our present (empirical) understanding of β and density limits, and ohmic confinement. If systems of $\lesssim 500$ MW(e) are desirable then L/H mode confinement is still adequate but approaches optimised ohmic confinement.

4 Advances in Exhaust Technology for Commercial Power Reactor in Mid 21st C

4.1 Introduction

Exhaust technology for a fusion power reactor must provide viable engineering for all phases of normal operation – start-up, stationary burn phase, shut-down – and for abnormal transient events such as disruptions and ELMs (Edge Localized Modes associated with the enhanced confinement H-mode) which expel large amounts of plasma and energy. In current reactor study this is attempted against a background of uncertain physics and limitations in materials. The objective here is to highlight areas where advancement is necessary, and to identify possible approaches either by further development along current lines of activity or by innovation, so that the technology can be applied to the problems of exhaust on a reactor with greater assurance.

In assessing the present state of divertor design, rather than discuss specific proposals in detail, a more general approach is adopted. A selection is made from the large number of tokamak next-step and power reactor studies which is fairly representative of the different types of reactor being considered. The selection and classification is as follows for D-T:

- **REFERENCE 2** Power reactor, medium field, medium aspect ratio, first stability regime (Cooke, this Review).
- **ITER** Next-step, medium field, medium aspect ratio (R/a_s), first stability regime.
- **ESECOM** Power reactor, medium field, medium aspect ratio, first stability regime, representative of lowest-cost D-T ESECOM variants (e.g. V-Li, V-Flibe).
- **ARIES-I** Power reactor, high field, large aspect ratio, first stability regime.
- **ANL/TPSS** Power reactor, medium field, large aspect ratio, second stability regime.
- **CSTR** Power reactor, low field, tight aspect ratio spherical torus type, direct conversion in divertor.
- **CFAR** Power reactor, high field, medium aspect ratio, high temperature, direct conversion of synchrotron radiation to increase overall power conversion efficiency.

For $D -^3 He$ we include:

- $V - D^3He$ Power reactor, high field, high temperature, direct conversion of synchrotron radiation (ESECOM).

For the present purposes, each of these variants is taken to have a poloidal divertor, although it was not specifically included in some (ESECOM). Indeed we note from recent results on DIII-D that the H-mode can be achieved with a limiter configuration and so a poloidal divertor may not be required at all for enhanced confinement. Basic parameters of the devices are given in Table I, along with estimates of the Bremmstrahlung (P_b) and synchrotron (P_s) radiation losses estimated using the following expressions (Generomak report):

$$P_b = 0.11 \ <n_e>^2 <T_e>^{1/2} \ Z_{eff} \ R_o \ ab \ \ (MW)$$

$$P_s = 0.25 \ <n_e> <T_e> B_o^2 \ Rab\phi \ \ (MW)$$

where

$$\phi = 1.56 \times 10^{-3} \ <T_e>^{1.1} \left(\frac{B_o}{<n_e> \bar{a}}\right)^{1/2} \ (1 - R_w)^{1/2}$$

with $Z_{eff} = 1.5$ and $R_w = 0.95$ (wall reflection coefficient for synchrotron radiation). The units of n and T are $10^{20}m^{-3}$ and keV. We note that some control can be exerted over the synchrotron emission by varying R_w. Subtracting P_b and P_s from the sum of the alpha-power P_α (and protons for $D -^3 He$) and the heating/current drive power, P_H, gives the maximum thermal transport power $P_\perp$ which must be handled by the divertor. Unlike most other studies we do not introduce arbitrary amounts of impurity radiation loss to reduce $P_\perp$.

A useful figure of merit is the power exhaust per unit toroidal length of the device, $P_\perp/2\pi R_o$. The comparison in Table I shows that CSTR has a value far above those of the other devices. Of the power reactors, ARIES-I and CFAR have low values as much of the power is transferred to the first wall in synchotron radiation. REFERENCE 2 also has a relatively high power per unit length.

The selection of materials suitable for use as divertor plates requires an assessment of performance both under plasma bombardment, producing erosion by sputtering, and under high steady and pulsed heat loads which lead to problems from thermal stress and recrystallization. In Section 3 the currently favoured materials are compared, and some possible developments and innovations are presented.

Divertors with lower performance would be required if a substantial fraction of the heating power in the reacting plasma were spread over the first wall in radiation. Two possible schemes for achiveing this, synchrotron radation from the bulk and edge radiation, are discussed in Section 3. The potential to improve the overall power conversion efficiency of future reactors by direct conversion, either using the properties of a diverted plasma, or by coupling synchrotron radiation through penetrations in the first wall into MHD convertors is also presented.

4.2 Present State of Divertor Design and Limits to Performance

4.2.1 Stationary Burn Phase

General Considerations

The three functions which a divertor must perform on a reactor are impurity control. power handling and helium removal. The critical issue of divertor technology, however, is erosion of divertor plate material during the burn phase, which has implications for impurity control and maintenance/replacement of the plates. The issue arises out of the many-keV energy per particle in the outward transport from the core plasma: it is given approximately by $P_\perp/\Gamma_\perp \sim 3 < T > \tau_p/\tau_e$. Typically $\Gamma_\perp$ has values in the range 10^{22} to 10^{23} particles per second.

The so-called High Recycling Regime (HRR) is thought to be the only credible operating scenario for a reactor divertor; the energy per particle is reduced by locally increasing the particle flux to the divertor targets by more than two orders of magnitude through additional ionization of neutral gas recycled from the targets. The energy per particle can be reduced to below the sputtering threshold for high-Z refractory metals, ~100eV, with the result of a long-life low-maintenance divertor target. The present discussion will therefore concentrate on this regime.

The Low Recycling Regime (LRR), which does not have this flow amplification, can be considered for a reactor if frequent target plate replacement is allowed. The high particle energies in the exhaust permit the use only of low-Z divertor plate materials, where the sputtering yield (including self-sputtering) can be kept sensibly below unity.

Detailed engineering designs for next step and power reactor divertors are broadly similar, and therefore, in order to provide a common point of comparison, all but one (CSTR) of the examples in Table I will be taken to have the same basic design with the following features.

- Double-null (DN), favoured over the single-null, because of the factor of 2 reduction in the power flow to each divertor.
- Expanded boundary: expansion of the flux surfaces and cross-field transport in the divertor channel increases the effective area of the power exhaust channel by about a factor of 6.
- More power flow to the outer divertor channel than to the inner channel, in the ratio 4:1 (ASDEX), although in the H-mode the ratio can be closer to 1:1.

- Solid divertor plates angled at about 17° to the flux surfaces on the outboard side to reduce the heat flux by a further factor of 3·4. Shallower angles appear to the impractical. On the inboard side the plates are allowed in most designs to be nearly perpendicular to the flux surface because of the lower heat flux.

- Sufficient space allowed in the main vessel between the separatrix and the first wall for a thick scrape-off layer to develop with $T_{e,i} \sim 10$eV to the wall, to reduce sputtering.

CSTR, in its long divertor channel varient, has a larger expansion, about 20, but angles closer to 45°, giving about a 30-fold reduction of power density in total.

4.2.2 High Recycling Regime Models

Although sophisticted 2-D plasma (e.g. BRAAMS code) and 3-D neutrals models (e.g. DEGAS) exist for the edge plasma and are being applied in increasing detail to reactor designs, it is difficult to extract from the results available sufficient information for scoping studies of different reactor types. The 1-D analytic model of Harrison and Harbour contains much of the useful physics of the HRR, and can be used for this purpose through its scalings on the important parameters. The model relies on:

- power flow in electron thermal conduction along the field with a lower T_e at the divertor plate than at the mid-plane, and $T_i \sim T_e$.

- nearly constant plasma pressure along the flow, with a low Mach-No. flow into the divertor.

- heat transmission across the sheath at the neutralizer with ions flowing at the local sound speed and an energy transmission of order $10kT_e$ per particle.

The 4 main equations which result from the analysis are as follows:

- Width of the power scrape-off layer at the outboard mid-plane:

$$\Delta_s = 9.1\ 10^{-3}\ (\chi_\perp n_s)^{7/9}\ R_o\ \left(\frac{a_s}{P_\perp}\right)^{5/9}\ q_\psi^{4/9}$$

- Power density at the outboard neutralizer/target plate:

$$P_t = 8.74\ \left(\frac{P_\perp^2}{\chi_\perp n_s}\right)^{7/9}\ \frac{1}{R_o^2\ a_s^{5/9} q_\psi^{4/9}}\ \frac{\sin\theta}{g}$$

- Electron temperature at the neutralizer plate:

$$T_t = 0.804 \left[\left(\frac{P_\perp}{a_s} \right)^{10/9} \frac{q_\psi^{1/9}}{\chi_\perp^{5/9}\, n_s^{14/9}\, R_o} \right]^2 \frac{1}{g^2}$$

- Electron temperature at the mid-plane ($T_s \gg T_t$):

$$T_s \simeq 3.09 \left[\frac{P_\perp\, R_o\, q_\psi^2}{a_s\, \Delta_s} \right]^{2/7}$$

The factor g represents the area expansion and sinθ the effect of the inclining the target plate, as described above.

The cross-field heat diffusivity $\chi_\perp$ is taken to be a constant as in 2-D modelling; the scaling would alter somewhat if a parametric variation of this were included. Positive dependencies on T_e, n_e, R_o would provide the greatest benefit in most cases: a key area for future work is to provide either well-documented empirical scalings for $\chi_\perp$ (and $D_\perp$) in appropriate plasma conditions or a well-substantiated theory for cross-field transport at the edge.

A necessary criterion for a T_e gradient to develop along the magnetic field is $\lambda_{ee} \ll L_T$, where λ_{ee} is the electron-electron mean free path and L_T is the gradient scale length, taken to be approximately equal to the connection length from the mid-plane to one divertor plate, $L_{||} \simeq \pi\ R_o\ q_\psi$. When this criterion is not satisfied, transitional or collisionless (LRR) flow will develop. The expression used for λ_{ee} is:

$$\lambda_{ee} \simeq 3.2\ 10^{-4}\ T_s^2/n_s$$

T_s (in eV) calculated above is used to check that the criterion is satisfied.

It should be noted that the experimental evidence for the existence of the HRR is patchy. ASDEX, DIII and PDX/PBX have demonstrated performance consistent with the HRR at low power, but as yet there is no convincing demonstration of its existence at high power, say on JET or DIII-D. An experimental verification of the difference in T_e between the mid-plane and the divertor plates remains to be performed.

High Recycling Regime Results

Table I shows the values calculated for the representative set of reactors using the expressions given above. In every case, $n_s =< n_e > /3$ was used for the mid-plane density at the separatrix, in common with the findings of many experiments, and $\chi_\perp = 4m^2s^{-1}$, as used in reactor edge studies. Note that with $\chi_\perp = 10m^2s^{-1}$ (more typical of the L-mode), Δ_s is increased by $\times 2$ and P_t, T_t and T_s are decreased by $\times 2$, $\times 2.76$ and $\times 1.2$ respectively. The main effects of these scalings for the representative reactors are as follows:

- Changing $\chi_\perp$ from $4m^2\ s^{-1}$ to $1\ m^2\ s^{-1}$, typical of the change from L-mode to H-mode produces a shorter power scrape-off scale-length, with increased power density, and higher target and mid-plane T_e values. Narrower scrape-off layers are observed experimentally in the H-mode, but power density and temperature changes are largely masked in present experiments by increases in edge or central radiation, or by ELMs.

- Large Δ_s, and low P_t, T_t and T_s are favoured by low $P_\perp$ and large R_o. This would appear to force CSTR into the LRR, even if 50% direct power conversion could be achieved in the divertor using Peng's scheme.

- The value of the mid-plane electron density is the critical factor in determining the other edge parameters (Harbour). For $n_s \lesssim 0.3 \times 10^{20}\ m^{-3}$ the edge will not be sufficiently collisional for the HRR to develop. As n_s is increased, the power flow, although collsional, is flux-limited by the neutralizer-plate plasma sheath, and T_s and T_t remain approximately equal. A further increase in n_s results in reduced target plate power density and electron temperature. For each case the precise value of n_s at which acceptably low T_t and P_t are achieved depends principally on the transported power. Values of n_s required to give T_t = 10eV are given in the table. Taking $< n_e > \simeq 3n_s$ would lead in some cases to violation of the density limit. In all cases the required $< n_e >$ values are probably too high for non-inductive current drive. Of those in the table, REFERENCE 2 and ANL/TPSS require the largest increases in edge density to achieve an acceptably small value of T_t. More radiation loss, reducing $P_\perp$, would ease this requirement.

- $V - D^3He$ comes closest to satisfying the requirements of collisionality. The others have too few collisions to be truly collisional and long-range electrons (Lackner), may begin to play a role. The high $P_\perp$ and low n_s of the REFERENCE 2 case appears to put it in the LRR. This area of work needs further theory and experiment.

These findings are confirmed by 2-D modelling, and broadly similar values are found for the edge parameters. Density scale-lengths at the mid-plane are generally long, as would be expected for low Mach No. exhaust. The more detailed models do, however, show surprising behaviour in the detailed flow pattern, with regions of reversed flow (from the divertor to the main chamber). This effect is particularly severe when T_t at the target plate on the separatrix flux surface is around 40 eV, and its persistence will have unfortunate consequences for the retention in the divertor of recycled helium and sputtered divertor plate material. Operation at higher or lower values of T_t is required to avoid it. Reversed flow has been observed on DITE, and it may play a role in the onset of the H-mode when the rate of density rise can be much greater than the fuelling rate.

4.2.3 Impurity Behaviour

The modelling of impurity behaviour is comparatively undeveloped; 2-D models are only now in the process of being put together. The main elements of the behaviour are, however, sputtering by ion bombardment, the trajectories of sputtered atoms, ionization of the atoms, redeposition, and impurity-ion transport along the magnetic field, with further ionization or recombination to change the charge-state. Sputtering and redeposition will be considered later, in Section 3.

In the HRR, ionization mean-free-paths for sputtered impurities are small, less than a few mm, and therefore the probability that a sputtered atom will reach the main plasma without being ionized is slight.

The transport of impurity ions along the magnetic field is governed, close to the neutralizer plate, by the electrostatic force due to the ambipolar electric field, and further away by a competition between friction with the background plasma ions and thermal forces between the impurity ions and the background plasma gradients, and hence towards the main plasma. Neuhauser has shown that the frictional force dominates when the following criterion is satisfied:

$$M > \lambda_{ii}/L_{T_i}$$

where M is the plasma flow Mach No., λ_{ii} is the hydrogen ion mean free path and L_{T_i} is the ion temperature-gradient scale length. Modelling shows that the background ion T_i gradient is generally steeper than the T_e gradient, giving a corresponding larger thermal force. Where impurity sources are localized near the divertor plates, backflow generally does not occur. Sources of impurity near the divertor entrance and in the main chamber are most likely to lead to backflow, and hence contamination of the main plasma. Much more work is required in the future to elucidate these effects, both by experiment and by modelling.

For impurity ions which are confined in the divertor, Shimada has developed a variation in the HRR, with Remote Radiative Cooling (RRC) of the electrons by impurity radiation in the divertor as well as by recycling. Radiation from the divertor is observed in experiments (ASDEX, JET, JT-60, DIII-D), and it goes a long way towards reducing the power loading on the plates by direct plasma bombardment, spreading it instead over the walls of the divertor chamber. Too much radiation can cause the onset of a thermal instability leading to disruption in extreme cases. Nevertheless, RRC is clearly a most promising avenue for future work, as this additional energy loss would ease the requirement for a high edge density in the HRR.

Helium Exhaust

The steady-state production of helium during the burn is about $1 \times 10^{21}\ s^{-1}$ for a device with 600MW of α-power, compared to a natural particle outflow of $\sim 5 \times 10^{22}\ s^{-1}$ from cross-field transport. Whether the helium ions escape into the SOL depends on the helium transport in the bulk plasma, a subject which is beyond the scope of the present assessment. These numbers show, however, that helium will represent $\sim 2\%$ of the particle flux into the divertor. The pumping requirements are determined therefore by the background hydrogen isotopes rather than the helium, with speeds in excess of 100 $m^3\ s^{-1}$ required to establish an acceptable neutral particle density in the divertor.

The impurity transport model (Neuhauser) discussed above predicts that the helium locally will be compressed in the plasma near to the divertor plates, increasing the percentage of helium locally in the flux to the divertor plates by up to 4 times, easing the pumping requirements by the corresponding amount. Nevertheless, to pump this quantity of gas requires large ducts through the blanket (with a corresponding loss of breeding capability) and pumps which will operate at high throat pressures, $\sim 10^{-2}$ mbar. These requirements are achievable with conventional technology, but at some expense.

Some metals, for example vanadium (as proposed for the ANL/TPSS limiter), have the capacity to trap helium preferentially. Fresh layers of getter would need to be deposited periodically in a short duct behind the divertor, reducing the amount of penetration of the blanket and the required capacity of mechanical pumps. Contamination of the divertor plates with a medium-z metal such as vanadium is undesirable because of its high radiation efficiency in the main plasma, and would have to be avoided. Nevertheless this sort of approach could be most fruitful for reactors.

4.2.4 Transient Phases

Start-up and Shut-down

Early during start-up, before the diverted equilibrium is established the plasma column will have to be retested on armour, probably on the inside wall. Some attention will have to be given to the material of the armour. If a refractory metal (e.g. W) is chosen for the divertor, then the choice of material for the armour becomes difficult. Metal armour has generally proven to be unsuitable in present-day large tokamaks; graphite has been more successful, but is not without problems (see Section 3). However, the effect of carbon deposits on tungsten may not be too severe, as sputtering and redeposition will probably move the carbon away from the region of highest heat flux.

Once the diverted configuration is established, sputtering of the plates will be a problem if the evolution of the edge density cannot be tailored to the power transported into the

SOL. The HRR will not develop if the density is too low. Operational scenarios which raise $< T_{e,i} >$ faster than $< n_e >$ will pose a problem here (e.g. non-inductive current ramp-up), and therefore the compatibility of these schemes with the HRR divertor needs further study. An acceptable level of erosion during start-up will have to be established.

Shut-down poses similar problems, but they will be alleviated if radiation cooling is used to reduce the plasma energy content.

Disruptions and ELMs

For a typical reactor ($\beta_T = 0.05$, $B_o = 5$T, R $= 5$m, a_s= 1.5m), the plasma energy content is 500MJ. During a disruption a significant fraction $\sim 50\%$ of this energy can be dumped in the SOL, and hence on the divertor plates. With typical flux surface expansions and target inclinations, the energy flux to the divertor plates will be in the range 5 to 20 $MJ.m^{-1}$. The characteristic time to dump this energy is$\sim 100\mu s$ and the instantaneous power density to the target plates will be some 4 orders of magnitude greater than that in the steady burn phase. Clearly the HRR will provide no help here, and the characteristic energy of the ions hitting the plates will be $< T >$. Erosion will be unavoidable. The main thrust of materials development must be, therefore, to find suitable materials to support the high surface heat flux and thermal shock caused by these events.

ELMs have similar dynamics to disruptions but with less energy being expelled at each event. In the steady H-mode with ELMs (DIII-D), typically there are several ELMs in an energy confinement time. Each ELM expels a small fraction ($\sim$ 10%) of the plasma stored energy and a similar fraction of the particles. While the average energy confinement time is reduced by a small amount ($\sim$ 30%) when compared with the ELM-free H-mode, the effect on particle and impurity confinement is dramatic, arresting the characteristic density rise and inhibiting the accumulation of impurities in the centre. However, from the point of view of the edge, operating in the H-mode with ELMs subjects the divertor plates to regular short pulses of energetic ($\sim$ keV) particles and it is not clear that erosion will be any less than in the LRR.

4.2.5 Alternative Schemes and Innovations

Ergodization

In the context of divertor physics, the main function of ergodization would be to increase $\chi_\perp$ at the separatrix, thereby increasing the energy scrape-off scale length and reducing the target power density and electron temperature. Increased transport at the separatrix would need to be achieved without affecting access to high confinement regimes which depend on the X-point magnetic geometry. Ergodization, when it has been tried on experiments (TEXT, TEXTOR), has not been truly random; discrete structures have been observed in the edge plasma. There is some doubt that sufficient 'true' randomization could be achieved to be of benefit in the way required in the SOL. Close-in coils will suffer radiation damage, requiring frequent replacement. Considerable work remains to be done to make this a viable option for a reactor.

4.2.6 Divertor Sweeping

By placing a poloidal trimming coil (normal copper) inside the blanket close to the target, the strike point of the scrape-off layer can be swept over the target, reducing the time-average heat flux, and possibly spreading the erosion and redeposition over a wider area. This is quite an attractive idea in principle, and recent studies have shown that movements of 10 cm at frequencies 0.1Hz are acceptable from the point of view of thermal stresses in the divertor plates. The main disadvantage seems to be the dissipation caused by the oscillating currents induced in the superconducting main coils, but nevertheless the idea deserves further pursuit. Oscillating the plates is also worthy of consideration.

4.3 Possible Improvements in Divertor Technology

4.3.1 Materials

General Considerations The ideal material for a divertor plate on a reactor would have:

- Low sputtering yields for the plasma temperatures in front of the plate.
- Durability to extreme surface heat loadings such as occur during disruptions or ELMs, as well as to the very high heat loadings of the stationary burn phase. A wide variation in the power loading can arise also from modulation of the flux surfaces by the TF coil ripple (especially for limiters), from ripple-trapped particles and from mechanical misalignment, given the smallness of the scrape-off scale lengths relative to the overall scale of the devices.
- Resistance to irradiation damage from neutron bombardment.

The first two requirements would appear to lead naturally to the choice of a high-Z refractory metal, were it not for uncertainties in the plasma physics of the edge, such as:

- The viability of the HRR to reduce the electron temperature at the plates, and its compatibility with other requirements such as current drive, density limit etc.

- Plasma conditions during start-up and shut-down.
- The effect of plasma flow reversal, or impurity transport up the temperature gradient, on the ability to retain the impurities in the divertor.

Currently, because of these uncertainties, it is believed that for the next step the choice of a high-Z material is unwise, with carbon based materials being favoured instead as being a more robust choice given the uncertainties of the plasma physics. For the LRR, high-Z materials are ruled out because of the high self-sputtering yield.

In the next section, the relevant parameters, and potential for improvement of candidate materials are described.

Assessment of Materials

- <u>CARBON</u>: At low surface temperatures, direct sputtering yields are modest for D, T, and He ions, but are high, ~ 1, for O ions and for self-sputtering. Threshold energies are around 30eV, and so sputtering is inevitable even at the lowest practicable electron temperatures. Chemical sputtering, by O or H isotopes, dominates between 700k and 1100k, and radiation-enhanced sputtering takes off above 1600k. The ideal surface temperature is 1200-1500k, but outside this range, total yields of order unity are possible if all of the ion species in the plasma are considered. Sublimation sets the ultimate limit on the surface temperature. Yields are relatively insensitive to the type of graphite. Measurments, supported by modelling, of erosion and redeposition on limiters have shown a removal of material from the region of highest heat flux to adjacent regions of lower heat flux. Little information is available so far on eroision and redeposition at divertor plates.

  Improvement in power handling is possible by using Carbon Fibre Composite materials. Adding a metal such as Beryllium and Tantalum to the composite with the aim of reducing radiation-enhanced sputtering at the higher operating temperatures has been considered. However, the potential of Carbon Fibre Composites, either loaded or unloaded, really depends on the properties of redeposited sputtered carbon on such a substrate, which at present is unknown.

- <u>BERYLLIUM</u>: Does not exhibit chemical or radiation-enhanced sputtering and removes oxygen by gettering. Total sputtering yields are expected to be slightly lower than those for carbon, with similar thresholds. It is high thermal conductivity, comparable to amorphous carbon and tungsten and at a much lower density than tungsten, is its main advantage, while its low melting-point is the main limitation.

- <u>TUNGSTEN</u>: Under high pulsed heat loadings, recrystallization without melting, surface melting and the formation of cracks can occur (Kny et al.). The power density threshold for melting is about 70MW.m^{-2}, while cracking and recyrystallization occur above 60MW.m^{-2}. For bulk temperatures above 300C, cracking does not occur, and therefore recrystallization remains the most serious thermo-mechanical problem.

Alloys, particularly alloys of rhenium with tungsten have higher recrystallization temperatures than pure metals, making them better able to support high heat loadings. The sputtering yield for rhenium is comparable to that for tungsten, but nothing is known about the properties of redeposited mixtures of the two metals. Nevertheless these alloys could make suitable target materials if low T at the target plate can be achieved in the HRR. Activiation will determine the choice of alloy metals.

Spheres, Liquid and Gas Targets

The use of a moving 'curtain' of spheres of solid material instead of integral carbon or metal structures could alleviate some of the problems of divertor targets. By an appropriate choice of speed, diameter and material properties, susbstantially higher average power loadings could be handled; sputtering and redeposition would tend to cancel each other out as particles move between high and low temperature regions of the exhaust channel. Unipolar arcing might also be eliminated (Hugill). Such a target would seem to be more suited to the LRR which is less geometry-dependent than the HRR. As gravity is used to form the curtain, it is hard to see how an upper-null could be accommodated. The power loading on the lower target is thereby doubled. Motion of charged spheres in E and B fields and the effects of floating, isolated spheres on potentials in the SOL need to be quantified.

These comments apply equally to a curtain of liquid metal droplets for which the additional problem of containment of the high vapour pressure of target material in the divertor chamber would have to be solved. From the engineering point of view a 'curtain' of liquid metal droplets (e.g. Li) would be preferable to solid spheres.

Flowing liquid metal films (e.g. Sn) on a solid base have also been proposed to give a continuously reviewed surface on the divertor target. The HRR is compatible with this technology, but the main weaknesses are the limited range of suitable materials, being mainly medium and high-z, and the lack of resistance of the film to disruptions and ELMs.

Extrapolating the HRR to even lower electron temperatures in the divertor could remove the need for a divertor target. Recombination is achieved in the plasma and the energy output is absorbed in the gas target. Maintaining the stability of system against pressure and thermal excursions poses the main problem, especially for disruptions and ELMs, but the concept is certainly worth pursuing especially if a long, closed divertor channel can be incorporated in the reactor design.

These 'innovations' all require substantial changes to incorporate them into reactor designs. Only the liquid metal film retains the 'base-line option' of an alternative solid target with the appropriate geometry. All of of the techniques require substantial development.

4.3.2 Radiation and Direct Conversion

Synchrotron Radiation

It was shown in Section 1 that infra-red synchrotron radiation emission from a tokamak-like plasma depends on $< T_e >^{2.1}$ and $B_\phi^{2.5}$, and consequently, by operating with high T_e and B_o the power to be handled by divertor can be much reduced. Limitations on τ_E dictate that the fraction of power radiated cannot be too large. A high energy density, is still required if the remaining transported power is to be handled by a divertor with a HRR.

Clearly the problems of a divertor in the burn phase on a reactor would be eased considerably if this form of radiation could be used to distribute the power over the first wall.

The synchrotron radiation can also be harnessed for direct conversion by guiding the radiation into a MHD generator. This has the potential for improving the overall energy conversion efficiency of the plant to > 50%. Among the types of MHD convertor, high-temperature, ceramic wall, seeded gas (Cs + Hg) and low-temperature, metal wall, liquid metal (Li or LiPb) convertors are worthy of consideration as well as a number of solid-state approaches. The cost of electricity from a plant with direct conversion could be reduced towards a level comparable with that from conventional nuclear power. Further cost reductions could be achieved in blanket-less reactor designs for aneutronic fusion, using $D -^3 H_e$, if the appropriate plasma conditions become attainable.

4.3.3 High Edge Radiation

Another method of reducing the required edge density is to reduce the energy transported into the SOL from the core plasma by radiating it close to the separatrix. Whether or not a highly-radiating edge can be made to work in a reactor-grade tokamak plasma depends largely on the cross-field impurity transport. Impurity accumulation can occur in the ELM-free quiescent H-mode, leading to a radiation collapse of the central temperature which should be avoided. Operation without impurity accumulation is observed in other confinement regimes. The ideal impurity would be medium-Z, radiating strongly near the edge, but fully stripped in the core. It would need to be deposited close to the separatrix for maximum efficiency, requiring beams or pellets.

Electrodynamic Conversion

Peng has proposed a scheme for CSTR with a LRR divertor which has the potential to recover 50% of the transported power by Direct Electrodynamic Conversion (DEC) in the divertor. The principle of the scheme is that hot ions ($\sim$ 1keV) in the SOL are channelled into a remote chamber, inside the TF system, and allowed to move towards the centre column, mirroring off the increasing toroidal field. If the bounce time of the collisionless mirroring ions is greater than the ($\nabla B + 1/R$) drift time, then the ions will drift out of the SOL in the divertor and can be collected. Collisional electrons are collected on a second plate with the ions which remain in the mirror loss-cone, and a current can be drawn between the two plates.

The requirement for a large mirror ratio makes this technology applicable only to tight aspect ratio devices; the anticipated conversion efficiency drops to less than 20% for a 'conventional' A=3 reactor divertor.

A drawback of the scheme in its requirement for a high-temperature exhaust with the attendant problems of erosion, particularly on the plate near the centre column which is near-perpendicular to the flux surfaces. Excursions in the DEC circuit, caused by disruptions or ELM-like behaviour, will also be a problem. The potential of this technique for highly-effective direct conversion cannot be ignored and further development is certainly worthwhile.

Electrostatic separation on the low-field side of a tokamak has been proposed for direct conversion, but as the method would appear to require a device like a bundle divertor to extract the plasma stream, it is unlikely to have any application to reactors.

4.4 Summary

An assessment of fusion reactor designs shows that, for the divertor:

- The 'conventional' approach has great difficulty in achieving a high recycling divertor unless operation at higher density and with larger radiation loss can be achieved.
- High field, high T_e approaches can satisfy these requirements. The enhanced synchrotron radiation loss, as well as reducing the transported power, can be harnessed for direct conversion, improving the overall efficiency of the device. This is of particular interest for aneutronic fusion using $D -^3 H_e$.
- Small aspect ratio devices will probably have a low recycling divertor requiring frequent plate maintenance. Direct conversion in the divertor maybe possible for these devices, improving the conversion efficiency.

Techniques such as ergodization and divertor plasma sweeping should improve the ability of the divertor to handle the high steady-state particle and heat loads.

Little improvement is expected in materials technology, the main constraint being sputtering of the surface, reducing the scope for the use of high heat-flux alloys. Low-Z is preferred for low-recycling applications.

If $T_e \sim 10$eV could be achieved in the plasma near the plates in a high recycling divertor, it would then be possible to use a longer-life high-Z refractory metal for the target. The main factor then affecting the life of the divertor plate is damage due to transient events such as disruptions.

Table 3

PROJECTED EDGE PARAMETERS FOR NEXT STEP AND POWER REACTORS

PARAMETER	SYMBOL/UNITS		REFERENCE 2	ITER	ESECOM	ARIES I	ANL/TPSS	CSTR		CFAR	$V - D^3\ He$
Major Radius	R_o	(m)	5.31	5.8	5.35	6	6	2.7	1.6	6	8.56
Minor Radius	a	(m)	1.4	2.2	1.35	1	1	1.5	1.33	1.2	2.37
Elongation	b/a		2.5	1.9	2.3	2	1.3	3	2.7	2.4	2.2
Axial Toroidal Field	B_o	(T)	6.22	5	4.31	12.4	4.4	4.77	0.6^d	10	10.12
Plasma Current	I	(MA)	16.6	22	14.6	9.5	4.0	42.6	20	20	60.2
Safety Factor	q		2.17	3.2	2.3	5	2.2	2.3	7	2.8	2.1
Volume Average Density	$< n_e >$	$(10^{20}\ m^{-3})$	1.45	1.0	2.1	1.8	2.33	1.1	2.1	1.1	2.55
Volume Average Electron Temperature	$< T_e >$	(keV)	20	10	10	20	17	15	20	43	40
Fusion Power	P_F	(MW)	3050	1000	2230	2400	1950	3043	2200	2750	3258
α-Power	P_α	(MW)	610	200	446	480	390	608	440	550	3204^b
Heating/Current Drive Power	P_H	(MW)	91	-	-	89	91	-	50	-	-
Bremsstrahlung Power	P_b	(MW)	40	28	51	29	29	14	26	27	717
Synchrotron Power	P_s	(MW)	95	19	10	345	15	15^e	1	380^a	1900^c
Transport Power	$P_\perp$	(MW)	566	153	385	195	437	579	463	143	587
Power/Unit Length	$P_\perp/2\pi R_o$	$(MW\ m^{-1})$	17	4.2	11.5	5.2	11.6	34	46	3.7	11
Separatrix Electron Density	n_s	$(10^{20} m^{-3})$	0.48	0.33	0.7	0.6	0.77	0.37	0.72	0.93	0.85
Mid-plane Power Scrape-off Scale Length	Δ_s	(mm)	5	12	9	14	6.9	3.8	4.0	10	16
Target-plate Peak Power Density	P_t	$(MW\ m^{-2})$	80	8	33	8.8	41	251	202	8.5	16
Mid-plane Separatrix Electron Temperature	T_s	(eV)	>450	107	139	178	>180	>258	240	118	122
Target-plate Electron Temperature	T_t	(eV)	450	35	68	52	180	258	105	57	11
Connection Length	$L_\parallel$	(m)	36	58	38	94	42	20	35	53	56
Electron-electron Mean-free-path	λ_{ee}	(m)	>135	11	8	16.9	>14.6	>57	25	12	5.6
Separatrix Electron Density for T_t = 10eV	n_s, 10	$(10^{20} m^{-3})$	1.62	0.49	1.3	1.02	2.0	1.05	1.53	0.65	0.88

a From CFAR Paper (31% of value given by formula)

b $p + \alpha$

c Scaled from CFAR

d Actual 1.6T including paramagnetic contribution, used to calculate P_s

e Does not include correction from paramagnetism (> 4)

5 Possible advantages of spin polarised nuclei in reactors

It is well known that the cross-sections of nuclear reactions depend on the spins of the interacting nuclei. In the case of D-T fusion two distinct polarisation modes have been considered. In the first, referred to as the C-mode, the spins of both the deuterons and tritons are aligned along the equilibrium magnetic field ($\underline{B}$). For perfectly polarised fuel this leads to a 50% increase in the fusion cross-section independent of the energy of the reacting nuclei and, hence, a 50% increase in the fusion rate coefficient. The reactivity enhancement drops rapidly with decreasing polarisation fraction; for 60% polarisation the enhancement is reduced to $\sim$ 10%. Unlike the unpolarised case it also leads to the fusion products being created with an anisotropic velocity space distribution $\propto \sin^2\theta$ (where θ is the angle between the velocity vector and the field line), ie preferentially perpendicular to the field. The second mode, referred to as the P-mode, the tritons are unpolarised while the deuterons are polarised with their spins perpendicular to $\underline{B}$. This leads to no enhancement of the cross-section compared with the unpolarised case but leads to an anisotropic distribution of the fusion products with a probability distribution $\propto (1 + 3\cos^2\theta)$ and, therefore, a peaking along the direction of $\underline{B}$.

The increased reactivity offered by C-mode polarisation would reduce the Lawson product (nτT) required to reach ignition and might improve reactor economics in a variety of ways depending on which of the reactor parameters are varied. With all machine and plasma parameters held constant the fusion power and wall loading increase linearly with the enhancement in the fusion rate s $= \sigma/\sigma_o$. If the increased wall loading can be handled without significant additional capital or operational cost this would reduce the cost of electricity. A study of a second stability regime reactor by Finn et al gave a reduction in the direct cost of electricity of $\sim$ 16% for a reactivity increase of 1.5. However, this saving would be eliminated if increased maintenance of the first wall was required. An alternative strategy would be to maintain the fusion power and wall loading constant while reducing $\beta^2 B^4$ in line with the reactivity enhancement. This would allow operation at lower β (which would reduce concerns about β-limits), at lower B or a combination of the two. At constant β and T_e the reduction in B ($\propto s^{-0.25}$), $n(\propto s^{-0.5})$ and I_p ($\propto s^{-1/2}$ assuming operation is at the β-limit) allows, amongst other things, (1) reduced magnet and structural material costs, (2) reduced non-inductive current-driver power ($P_{CD} \propto nI_p \propto s^{-0.75}$) and (3) reduced plasma energy content (W $\propto s^{-0.5}$) and hence reduced material ablation resulting from disruptions. The reduction in P_{CD} could be further improved to $P_{CD} \propto s^{-1}$ by maintaining both β and B constant but reducing the density by a factor of s. Studies by Finn et al also indicate that, at constant fusion power, operation using C-mode polarisation leads to a modest ($\sim$ 5%) increase in the wall life due to reduced neutron damage. The only potential disadvantage of the C-mode polarisation is that the fraction of trapped α-particles will be increased due to the angular weighting factor which will thereby increase the fraction of α-particles lost by a factor of $\sim s$. This should not be a significant concern except in low current (eg second stability) or high magnetic field ripple designs.

The P-mode does not increase the reactivity and its only potential advantage is that it produces an angular distribution of α-particles and neutrons peaked along $\underline{B}$. This would reduce the α-particle loss but as noted above this will probably not be a significant concern. The peaking of the neutron distribution along field lines somewhat reduces the fast neutron flux on the central column of the torus (which might make tight aspect ratio designs easier to implement) but increases the flux on the outer wall. Studies by Finn et al indicate that the increased loading on the outside will lead to reduced wall lifetime and higher reactor cost.

The major issues concerning the use of spin polarised fuel in reactors concern the production and maintenance of a plasma with the high levels of polarisation required. The generation of polarised fuel in sufficient quantities for injection into a reactor does not appear to be a problem since several methods are available, none of which appear to have a significant energy demand. However, technical improvements are needed to increase both the production rate and level of polarisation above those already achieved. Similarly, injection of polarised fuel into the plasma using either pellets or beams does not appear to pose insurmountable problems. The main concern is that depolarisation of the polarised fuel may occur in the relatively long ($\sim$ 10s of seconds) residence time of the fuel in a reactor. Theoretical work indicates that only two depolarisation mechanisms are significant. Firstly, spin depolarisation will occur if magnetic fluctuations (from either electromagnetic waves or plasma instabilities) with amplitudes significantly above the thermal level are present at frequencies close to the precession frequencies of either D or T nuclei. If the magnetic fluctuation amplitudes in reactors are similar to those seen in L-mode operation of present machines, depolarisation by instabilities might be a concern. However, high confinement regimes (which are probably of more relevance to reactors) often have significantly lower magnetic fluctuation amplitudes than L-mode plasmas and hence reduced capability to depolarise the fuel. Velocity space instabilities with $\omega \sim \omega_{ci}$ (driven for example by the anisotropic α-particle distributions arising from the use of polarised fuel) might also pose problems. Similarly, the use of low frequency ($\omega \sim \omega_{ci}$) heating or current-drive schemes may be precluded. Secondly, depolarisation of edge recycling atoms can occur rapidly (within milliseconds) in metallic plasma facing components. This would preclude the use of metallic divertor targets. The time for depolarisation to occur in non-metals is in the range 0.1-10s provided that paramagnetic impurities are kept at a low level ($\lesssim$ 10ppm). This may be sufficiently long to permit their use in a reactor using spin polarised fuel but detailed studies are still required.

Fuel polarisation also offers improvement in the fusion rate coefficient by up to $\sim$ 50%

for D-^{3}He systems. This enhancement could be used to ease significantly the technical requirements of a reactor in the same ways as was outlined above for C-mode polarisation of D-T fuel.