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1 Executive Summary

There are two major sections to this report. The first section of the report, which you are now reading, is an executive summary of the work done this year. For each task, the major results are condensed for the reader's convenience. The major result of each memo, report or presentation is summarized briefly in this section. The second section of the report is a collection of appendices containing reports, memos, and presentations written this year. Here, the interested reader can investigate any topic discussed in the summary in more detail. The documentation is presented in chronological order, and we would like to note that the content of later documents may supersede that of earlier ones.

The summaries are divided into sections, corresponding to the tasks outlined in the original proposal for the work. These sections are:

- MUMAK Code Development and Application
- Alfvén Wave Stability Problem
- TETRA Systems Code Development and Application
- Lower Hybrid Heating and Current Drive
- Advanced Blanket Modeling

We think that the work accomplished this year is valuable to the proper definition of ITER. A number of papers were presented at APS, ANS, and Sherwood meetings. The ANS papers will be published in a proceedings in a volume of George Miley's fusion journal, *Fusion Technology*. A paper on the concept of SOPCONS will be submitted to *Nuclear Fusion*. Work TRW has performed illustrating the impact that different divertor heat flux models (Barr versus Harrison/Kukushkin) have on ITER performance is important in retaining steady-state current drive operation as an ITER option.

1.1 MUMAK Code Development and Application

Power for ignition even for a steady-state machine is an important consideration. We addressed this issue in a memo dated February 2, 1988

(LLNL-ITER-88-026). The machine size was $R_0 = 4.04$ m, $a = 1.41$ m, $I_p = 16.4$ MA. For this case, the required heating power to make the transition from an ohmically heated equilibrium to ignition was estimated to be approximately 30 MW. This power could be supplied by lower hybrid heating, electron cyclotron resonance heating, ion cyclotron resonance heating, or neutral beam injection. From a comparison of these methods, we conclude that ECRH and NBI are the preferred heating methods. Note that to ignite the current baseline of $R_0 = 5.5$ m, $a = 1.8$ m at higher current could require more power to ignite, but because the confinement improves with size, the power increase would not be proportional to the volume increase. Additional MUMAK calculations will be necessary to calculate the power for the larger size taking into account these competing effects.

Choosing a steady state operating scenario for ITER is an important task. LLNL and TRW have developed an important tool for comparing different operating points called the Steady-state OPERating CONtours (SOP-CONS). These are contour plots of various plasma quantities (e. g. fusion power, wall loading, beam energy) as a function of plasma current and Troyon coefficient. A memo was written (LLNL-ITER-88-034) which addresses the choice of an operating point. A presentation was given at PPPL on this topic in June. ITER will have at least two basic operating scenarios: ignited burn with purely Ohmic current drive and non-inductively driven steady state operation. In the latter case, the operating scenario must satisfy a number of operational conditions, most prominently $Q > 5$, the average neutron wall load > 1 MW/m², as well as the recognized constraints on the physics parameters. The density must be less than the maximum density to avoid disruptions. Ideal MHD theory sets an upper limit on the plasma beta and a lower limit on the edge safety factor. The energy confinement time is related to plasma conditions through some scaling law. The power required for current drive is found from an appropriate scaling law. Part of the current may be available from neoclassical (bootstrap) effects. The memo outlines some possible operating points, and indicates that there is some freedom in choosing the point based on what constraints are considered important. Some important conclusions of the operating point work are:

- Total current drive power requirement is nearly constant for devices with major radii $R_0 = 4, 5,$ and 6 m, provided that Alfvén instabili-

ties do not severely degrade NBI current drive efficiency.

- Optimized steady state operation occurs with enhanced (diverted) L-mode energy confinement.
- Steady state operation satisfying the ITER goals is possible with a density well below the density limit, beta below the beta limit, and current below the maximum value (q limit).

TRW was involved in the study of the thermal stability of the steady state operating points for ITER using POPCON analysis, which looks at contours of required auxiliary power in density- temperature space. The work is documented in memo LLNL-ITER-88- 049. We found that for full H-mode energy confinement, the current drive efficiencies are sufficiently low that the sum of the power required to sustain the current plus the alpha power exceeds the total plasma losses for all values of density and temperature below the beta limit. The plasma is not in thermal equilibrium under these conditions. For lower values of the H- mode enhancement factor, thermal equilibrium windows in density temperature operating space exist. For "enhanced L-mode" confinement ($H=1.5$), regions in the thermal equilibrium windows are identified which are stable to temperature excursions. A further reduction in H to the L-mode confinement ($H=1.0$) results in a large external heating power requirement.

A possible step on the road to ITER may be the Steady State Tokamak Experiment (STE). STE is proposed to test some of the important steady-state and superconducting issues which will have an impact on ITER. TRW was involved in estimating the performance of this machine, and the work appears in LLNL-UCID-21404 and General Atomics document GA-A19305. In this work, a range of tokamak parameters were examined and the costs and benefits of varying the size was considered. The critical physics parameters examined by MUMAK are the scaling of energy confinement time and current drive efficiency. Sufficient power density at the plasma surface is also a requirement. An important consideration is flexibility, to allow the variation of plasma shape, and to permit possible upgrades in field, current, and power, subject to cost effectiveness.

In July, questions were raised in Garching about the viability of steady-state non-inductive current driven scenarios. The confinement scaling law used for the comparison with ignited scenarios was the JAERI scaling.

These concerns motivated us to perform a study looking at ITER steady state operating space for a set of three confinement scaling laws. The work is summarized in LLNL-ITER-88-053. The primary conclusion of the work is that attractive current-driven machines can be defined which obey the latest ITER rules, if either Goldston L-mode of 1.5 or Rebut-Lallia L-mode confinement applies to ITER. If L-mode JAERI scaling with $H=1.5$ prevails, the performance of a steady state ITER must tolerate a large current drive power (≥ 150 MW) for machines providing a reasonably high wall loading ≈ 1 MW/m². If full H-mode ($H=2.0$) is postulated for JAERI scaling, the performance improves dramatically, to the point where it is comparable to the other two laws. It should also be noted that using a model by Bill Barr for the peak divertor heat flux, the ITER rules of $\dot{H}_{div} \leq 5$ MW/m² can be nearly satisfied. This is in sharp contrast with the heat flux predicted by the Harrison model, currently the model being used by the Garching divertor team. The Harrison model predicts a considerably higher peak divertor heat flux. Barr's model has been benchmarked with D-III-D data and is in good agreement. To the best of our knowledge, the Harrison model has only been benchmarked to the Braams computational code.

We did some additional work this year aimed at challenging the divertor heat flux model of Harrison. The details of this work can be found in the memo LLNL-ITER-88-055. At a Garching workshop, statements were made that to keep the divertor heat flux below the limits imposed (≤ 5 MW/m²) the plasma gain factor Q would have to exceed a certain number. Using the Harrison model, this Q_{min} was actually quite high, approaching 10 at a neutron wall loading of 1 MW/m². We looked at this question using the Barr model (having been benchmarked to experimental data), and discovered that the Q_{min} is quite a bit lower in the regime where the limit is based on the peak divertor heat flux. This fact suggests that the negative statements made at Garching about steady state operation need to be revised. This sent the strong message that before any programmatic and/or configurational decisions are finalized, these divertor heat flux models need to be reviewed. An improved communication between the divertor groups was suggested.

As a review of the work on an extrapolation of ITER to a reactor, a contributed paper to the Salt Lake City ANS was written. In this paper, the physics and engineering guidelines for the ITER device are shown to lead to viable and attractive operating points for a steady state tokamak

power reactor. Non-inductive current drive is provided in steady state by high energy neutral beam injection in the plasma core, lower hybrid slow waves in the outer regions of the plasma and bootstrap current. Plasma gain Q in excess of 20 and average neutron wall loading of 2 MW/m^2 are predicted in a device with major radius $R_0 = 7.5 \text{ m}$ and minor radius, $a = 2.8 \text{ m}$.

1.2 Alfvén Wave Stability Problem

The baseline for ITER uses high energy neutral beam injection to drive a steady-state current. Beam penetration and standard current drive efficiency considerations lead us to consider energies in the range of 1-3 MeV. Apart from the ambitious extrapolation from present source and accelerator technologies, there is also concern that the high energy beams with speeds above the Alfvén speed ($V_A = \frac{B}{\sqrt{4\pi m_i n_i}}$) will excite normal modes of the plasma which could, through enhanced slowing down of the beam, degrade the current drive efficiency. In ITER, the Alfvén energy near the plasma center is about 300 keV.

To answer these questions about the stability of the shear and compressional Alfvén waves and their ultimate reduction in the current drive efficiency, a two dimensional quasilinear code was written and named the Simplified Fokker-Planck (SFP) code. The SFP code solves the appropriate kinetic equations in velocity space for the ion distribution function heated by energetic injected ions. Through the introduction of terms describing the interaction between hot ions and turbulence excited by the instability, the change in the distribution function due to the turbulence can be calculated. Along with some flattening of the distribution function in the vicinity of the turbulence, the ions are found to slow down more rapidly than they would if the instability were absent. Using the beam injection parameters appropriate for ITER, we find that the hot ion current produced by the beam is reduced by about 25% as compared to the hot ion current calculated from classical collisions-only theory.

The code was also applied to some data from the T-11 tokamak in the USSR, with some interesting results. This will be discussed later in this section. Next year, we want to compare the code results with some D-III-D data. We will be working with Arthur Molvik of LLNL on the interpretation of the charge exchange flux data coming from D-III-D.

The first memo we wrote (LLNL-ITER-88-012) dealt with the definition of the problem, the review of the relevant literature, and some analytic calculations of the growth rates. The conclusions of that memo will now be discussed. The numbers were taken directly from the memo, and they have changed somewhat with additional improvements in our calculational ability¹. The important thing to note is that the qualitative stability information has not changed.

For the compressional mode, the perpendicular width of the distribution function is an important parameter in determining the degree of instability of the mode. For the case of parallel injection of a beam above the energy where pitch angle scattering is important, the angular width of the source is important. Narrow widths are claimed to result in a more stable system, according to H. Berk, et. al. . For injection at an angle, as is the case for ITER, the perpendicular width of the distribution is more like $v_b \sin \theta_{inj}$, where v_b is the beam speed, and θ_{inj} is the injection angle. For ITER, using this width, the minimum beam density for the compressional mode to become unstable is about a factor of 25 lower than the beam density in the baseline. The growth rate of the mode is approximately 2000 sec^{-1} . (This growth rate has increased somewhat from these earlier estimates, to about 2×10^5)

The shear mode also appears to be unstable for ITER. For a beam distribution plateau width which is consistent with a Fokker-Planck calculation using the CQL code, the density threshold for this mode is a factor of 2.6 lower than the beam density for ITER. The growth rate is quite large, on the order of one-tenth the real part of the frequency, $\gamma \approx 2 \times 10^5 \text{ sec}^{-1}$.

We condensed this memo into a smaller package and submitted it to be included into the ITER workshop notes at the June meeting.

Once the code was operating reasonably well, a poster session was presented at the Sherwood Theoretical Meeting in Gatlinburg, Tennessee this year. The conclusions and recommendations presented there are:

- Preliminary results indicate that the baseline ITER is unstable to both the shear and compressional Alfvén modes.
- Quasilinear relaxation of the distribution due to these waves reduces the current drive efficiency by a factor of 2. (Note: Later calculations

¹The memo was written before the SFP code was developed

performed with more realistic source widths result in only a 25 % reduction in the current drive efficiency.)

- Careful choice of operating parameters can improve the stability picture.
- Results are sensitive to the details of the source of energetic particles. (See item 2 above.)
- Experiments on D-III-D are very important in testing the theoretical predictions.
- Experiments on T-11 suggest that Alfvén waves won't be important, according to the experimentalists. (later work to be reported below give some indication that the experimental results do indeed suggest that there may have been instabilities present)
- Code simulation of T-11 suggest the compressional mode is stable, and the shear mode is unstable.
- Examination of the global (radially dependent) shear mode is also important to consider, and will be considered in future work.

Since that work for the Sherwood meeting was completed, we did some additional examination of the T-11 data. The results of this work is summarized in LLNL-ITER-88-042. This memo discusses the neutral beam modeling of the T-11 tokamak and implications for the Alfvén waves in ITER. The conclusions from that memo are summarized below:

- The compressional branch of the mode is stable in T-11.
- The shear branch is unstable, and the effect of the ensuing turbulence is to reduce the density of the hot ion plateau, *not* change the basic shape of the high energy tail.
- From the calculation, the presence of Alfvén instabilities in a device with injection of both super and sub-Alfvénic particles can be determined by comparing the plateau levels produced by each class of particle.

- The Alfvén waves deplete the plateau containing the super-Alfvénic particles, and leave the plateau levels from the slower particles largely unchanged. The T-11 data compared with the classical theory from the Soviets normalized to the bumps produced by the sub-Alfvénic particles also demonstrates this principle.
- The primary effect observed is to reduce the density of ions above E_A , but does *not* eliminate neutral beam current drive from consideration for the steady-state ITER.
- A methodology for detecting the phenomenon is outlined in the memo. Further experiments on large Tokamaks (e. g. D-III-D) are essential.

Soon after the T-11 memo was written, a presentation was given by R.S. Devoto at the Current Drive and Heating Workshop at Garching 14 June 1988. TRW prepared the presentation and it is included in the Appendices of this report. The reception of the material was in general favorable, and a need for continued work on this problem was widely recognized. Additional material faxed to Devoto during the meeting from Campbell substantiating some of the calculations have been included. This was the first time the realistic source parameters were used in the calculations. This turned out to be a stabilizing effect.

After the Garching meeting Devoto brought information home about a stability calculation performed by Stepanov. Stepanov pointed out that another mode may be important with frequency near the ion cyclotron frequency. He made estimates of growth rates which we feel are too high. A memo was written "Comments on Soviet Alfvén Wave Stability Calculation" (no ITER number) discussing aspects of the problem.

The stability calculation presented by K.N. Stepanov at the Current Drive and Heating Workshop indicated that a strong instability could exist in ITER which might preclude the use of neutral beams for current drive. This conclusion was arrived at by a global linear stability analysis which predicted very strong growth of a mode with a frequency just below ω_{ci} .

The claim made in the memo TRW produced is that the model distribution function he used, which has zero width in pitch angle, results in pessimistic answers. We have evaluated some of the pertinent terms in the

stability theory for a realistic ITER beam driven distribution and conclude that the growth rates will be lower than what Stepanov indicates.

Regardless of the size of the linear growth rate, a quasilinear calculation is necessary to determine the final effect on the beam heated distribution². We have the capability, after relatively small code modifications, to look at this mode and assess its effect on current drive.

We have also submitted an Abstract for a poster session to be presented at the American Physical Society, Division of Plasma Physics Meeting to be held October 31- November 4, 1988 in Hollywood Florida. This abstract is included in the Alfvén Wave Appendix.

At a certain point in the studies this year, Bill Nevins asked about the reasons why the turbulence should enhance the slowing down of the beam injected particles (hence why should there be any reduction in the current drive efficiency). His hope was that before the waves could do appreciable slowing down, the distribution function would flatten, thereby turning off the slowing down. To us, that question was equivalent to saying why should there be any hot tail buildup in the electron distribution function under the influence of externally applied lower hybrid waves. Without a hot tail buildup in one direction, there would be no lower hybrid current drive. Experimental evidence supports its existence. The mechanism for the buildup in the lower hybrid case and the slowing down in the beam instability problem is the same, just the shape of the original distribution function is different, which makes the answer the opposite sign (e. g. enhancement versus reduction in the hot density).

To illustrate some of these ideas, we wrote a memo "Why Does Alfvén Wave Turbulence Reduce the Current Drive Efficiency?" (no ITER number). The crux of the argument in the memo is now presented.

While it is true that some flattening of the distribution will occur, it cannot occur without some slowing down of the hot beam injected particles. The essence of the argument is that for a given source of hot injected ions, the slowing down flux must be the same regardless of whether it is due to Coulomb or anomalous processes. The Coulomb slowing down flux is primarily advective, meaning it is proportional to the distribution function. When an additional flux is introduced due to the Alfvén instability, (which can manifest itself even at low turbulence levels) the Coulomb flux must

²Stepanov did not do this

decrease. The only way for this to happen, recalling the advective nature of the operator, is for the local distribution function to decrease.

The result derived here also applies to electron cyclotron and lower hybrid heating and current drive, although in those cases the tendency is to *enhance* the hot tail density. This is because the distribution (Maxwellian) which is being acted on by the waves initially has a negative velocity space derivative, the flow of particles is then to higher energy.

1.3 TETRA Systems Code Development and Application

During FY1988, important features have been included in the TETRA systems/optimization/costing code. Studies of different value functions (other than cost of electricity), assessments of uncertainties in confinement scaling laws translating to uncertainty of ignition, and many different design studies to assess the impact of assumptions on capital cost and size were performed. The viewgraphs of several meetings where results were presented are included in the TETRA Appendix for the interested reader to examine.

Benchmarking the TETRA physics modules with the MUMAK code was also performed, and after several iterations of changing the TETRA models, we were able to achieve quite good agreement. This benchmarking is summarized in a chart in the MUMAK Appendix in this report.

The most notable improvement is the ability of TETRA to examine machines with multiple missions, the so-called "multi-modal" operation. The most common application of the "bi-modal" mode is to seek optimized machines which can succeed in both a steady state current drive and ignition application.

A paper will be presented at the ANS Salt Lake City Meeting during the week of October 9, 1988. In the paper, we discuss the expansion of TETRA to contain the option of examining machine designs capable of "multi-modal" operations. The conceptual ITER is an example of a "bi-modal" operating machine design in that it must be capable of both pulsed-ignited and steady-state operations. This new version of TETRA allows for the optimization of a "Figure-of-Merit" which may involve several operating scenarios. We obtain for ITER minimum cost designs which are capable

of igniting when operated in the pulsed mode as well as have reasonable $Q > 5$ when operated in steady-state.

1.4 Lower Hybrid Heating and Current Drive

Lower hybrid wave interaction with plasmas is an important topic in the steady state ITER scenarios. This year, we have produced two studies that have relevance in this area. The first is a survey of the experimental database (mostly after 1985) for lower hybrid current drive and heating. This complete report is included in the Lower Hybrid Appendix. The second study was performed using the ray-tracing Fokker-Planck code developed by Paul Bonoli at MIT.

With respect to the experimental database, the proposed ITER design may rely heavily on lower hybrid waves to provide heating, current drive, MHD stabilization through current profile modification and transformer recharging. The paper presents a detailed survey of recent LH experimental results from PLT, Alcator C, ASDEX, Petula-B, FT, and JT-60. Current drive and heating efficiencies are given, as well as regimes where sawteeth and $m=1$ and 2 oscillations are stabilized. In addition, in ASDEX and JT-60, LH waves in combination with neutral beams (a possible ITER scenario) experiments are examined. Finally, the current drive efficiency for ITER is obtained by extrapolating from the LHCD database. Assuming 12MW of LH power, approximately 4.5 - 5.0 MA of current could be driven in ITER. However, the high density ($n_e = 8 \times 10^{19} \text{ m}^{-3}$) and high temperature ($T_e = 21 \text{ keV}$) will preclude wave penetration to the center. Assuming a narrow $N_{||}$ spectrum ($1.2 \leq N_{||} \leq 2$) the LH waves should be absorbed within the outer half of the plasma.

We performed some studies with the Bonoli code this year. The results of this second study was presented at the FEDC at ORNL in March 1988. Issues examined with the Bonoli code included:

- Partial penetration into the plasma
 1. Limits total lower hybrid (LH) driven current.
 2. Constrains contribution of LH current to the total current density profile.
- Damping on fast alpha particles

1. Reduces the effective power damped on electrons (current drive) per unit power launched.
- Narrow $N_{||}$ spectrum requirement
 1. Large antenna with a very large number of waveguide slots is required.

From this calculation, we have determined some lower hybrid parameters for ITER. These can be summarized below:

- Frequency: $\geq 8 \text{ GHz}$
 1. Avoids damping on fast alphas
 2. Avoids parametric instabilities at plasma edge
 3. Data will be available on 8 GHz operation of a LH system from the FTU experiment this year
- Wave Spectrum: as narrow as possible
 1. Maximum penetration into the plasma
 2. Tailoring of the current profile by choice of $N_{||,0}$
- Launcher location: outboard midplane
 1. Assuming double-null divertor configuration
 2. For single-null configurations, bottom launch may improve penetration

The use of the Bonoli code and the interface of it to the ACCOME beam deposition code and the SELENE MHD equilibrium code will continue in FY1989.

1.5 Advanced Blanket Modeling

To improve the economics of a reactor extrapolation of ITER, some innovation in the area of blanket design and balance of plant would be desirable. In FY1987, considerable work was performed on defining the Magnetohydrodynamic generators and microwave superheaters for the CFAR advanced

blanket concept. During FY1988, much of this work was curtailed in favor of more work for the directly related ITER physics and system issues. However, enough work was done this year by TRW in the area of the MHD generator design that small, attractive generators were defined which interfaced nicely with the rest of the CFAR configuration. We contributed to an invited talk which was prepared primarily by Mike Hoffman at UC Davis discussing the complete concept and models for the CFAR system. It will be presented at the ANS Meeting at Salt Lake City, the week of October 9, 1988. A copy of the Abstract and Introduction to this paper are included in this document for the interested reader.

A MUMAK Code Development and Application

MEMORANDUM

To: ITER Physics Distribution

From: R. S. Devoto, M. E. Fenstermacher

Subject: Choosing a Steady State Scenario-Part I: Using SOPCON's

Introduction

ITER will have at least two basic operating scenarios: ignited burn with purely Ohmic current drive and non-inductively driven steady state operation. In the latter case, the operating scenario must satisfy a number of operational conditions, most prominently $Q > 5$ and average neutron wall load $> 1 \text{ MW/m}^2$, as well as recognized constraints on the physics parameters. The density must be less than a maximum density to avoid disruptions. Ideal MHD theory sets an upper limit on the plasma beta and a lower limit on the edge safety factor. The energy confinement time is related to plasma conditions through some scaling law. The power required for current drive is found from an appropriate scaling law. In addition, part of the current may be available from neoclassical (bootstrap) effects. In this memorandum we suggest a way to examine operating points satisfying the above constraints from the range of possible conditions.

In the next section we give typical expressions for the operating-point constraints. In the third section we describe Steady state Operating COntours, or SOPCON's, and show how they can be used to help select a range of operating scenarios. In a subsequent memorandum (Part II) we will use SOPCON's to select possible operating points for ITER designs of increasing major radii as well as a possible steady state reactor producing approximately 1000 MW of electric power.

Constraining Relations

A fundamental limitation on plasma parameters is the critical beta beyond which the plasma is expected to become unstable to ideal ballooning or kink modes. This limit is conveniently expressed in terms of the Troyon formula [1]

$$\beta \leq \beta_T = C_T I_p / a B_0 \quad (1)$$

where I_p is the plasma current in MA, a is the minor radius in m, B_0 is the toroidal field at the geometric major radius, R_0 , and C_T is a coefficient, typically 3 %, which sets the upper limit on

β . In addition, studies of ideal kink stability show that the plasma is likely to be unstable to $n = 1$ kinks if the edge safety factor q_a drops below about 3.0 [2,3]. The safety factor is given by

$$q_a = B_0 R_0 \int (ds / R^2 B_p) \quad (2)$$

with B_p the local poloidal field, and R the major radial coordinate. The limit on q_a is typically applied where the poloidal flux is 95% of the total flux enclosed by the separatrix. We thus can state the limit as $q_{95} > 3.0$. Since it is necessary to compute the 2-D MHD equilibrium to find the value of q_{95} , it is desirable to have a simple formula for safety factor. We can use the form

$$q_a = \frac{5 a^2 B_0}{R_0 I_p} \frac{1 + \kappa_{95}^2 (1 + 2\delta_{95}^2 - 1.2\delta_{95}^3)}{2} \quad (3)$$

where κ_{95} is the elongation and δ_{95} is the triangularity of the 95% flux surface. Since q_a is typically less than q_{95} , we should take our limit lower than 3.0, but we do not yet have sufficient information to choose a precise number. There are additional conditions on κ_{95} and δ_{95} , typically $\kappa_{95} < 2.5$ and larger δ_{95} required at larger κ ($\delta_{95} \approx 0.5$ at $\kappa_{95} = 2.5$ [3]).

For the density limit we use the Greenwald limit [4], given by

$$\bar{n}_e \leq n_{Gr} = f_{Gr} \kappa_{95} I_p / A_p \quad (4)$$

where $\bar{n}_e$ is the line-averaged density and A_p is the plasma cross sectional area and the coefficient f_{Gr} is taken as 0.75 to reduce the likelihood of density-induced disruptions. In practice, this limit should seldom be approached in steady state operation.

A variety of scaling laws are possible for energy confinement time. An oft-employed relation is the Kaye-Goldston form [5]

$$\tau_{KG} = 0.055 I_p^{1.24} R_0^{1.65} \kappa_{95}^{0.28} \bar{n}_e^{0.26} \cdot P_h^{-0.58} a^{-0.49} B_0^{0.09} (A_i/1.5)^{0.5} \quad (5)$$

where A_i is the average plasma mass, and the total heating power P_h is the sum of the current-drive power and the alpha power less the radiated power.

An alternate form is due to Goldston [6]

$$\tau_c = 0.037 I_p R_0^{1.75} \kappa_{95}^{0.5} p_n^{0.5} a^{-0.37} (A_1/1.5)^{0.5}$$

(6)

It is widely believed that the energy-confinement time cannot exceed the Ohmic, or neo-Alcator time given by [7]

$$\tau_{NA} = 0.0625 \bar{n}_e a R_0^2 q. \quad (7)$$

In practice, τ_{NA} is always greater than τ_{KG} or τ_c under predicted steady state conditions.

Both the Kaye-Goldston and Goldston scalings were developed for L-mode discharges in limiter plasmas. It was recently noticed that L-mode discharges in divertor plasmas in JET [8] displayed somewhat better confinement time, perhaps as much as 50% higher than the value predicted by τ_c . For these studies we therefore compute τ_E by multiplying the predictions from Eqs. (5) and (6) by a constant H in the range 1.2-1.5.

At present, the likely candidates for non-inductive current drive are lower hybrid (LH) slow waves in the outer region of the plasma, high-energy neutral beams (NB) in the core, and neoclassical (bootstrap) current. The LH waves are limited to the outer region because the choice of $N_{||}$ dictated by the accessibility condition means that the waves are strongly Landau damped on electrons in the outer plasma region where the temperature reaches 10-15 keV [9]. The accessibility condition is given by

$$N_{||} \geq N_{||ac} = \frac{\omega_{pe}}{\omega_{ce}} + \left(1 + \frac{\omega_{pe}^2}{\omega_{ce}^2} - \frac{\omega_{pi}^2}{\omega_o^2}\right)^{1/2} \quad (8)$$

where ω_{pj} and ω_{cj} are the local plasma and cyclotron frequencies of species j and ω_o is the frequency of the launched waves. The condition for strong electron Landau damping can be written approximately as

$$N_{||} \leq N_{||}^{ELD} = 7 / T_e^H(r) \quad (9)$$

In our calculations of the power required for current drive, we choose $N_{||}$ at the radius of maximum penetration into the plasma, r_{max} , to satisfy both of these conditions. The required power is reported in the form of a current-drive figure of merit

$$\eta_{lh} = \frac{n_{e20}(r_{max}) I_{lh} R_0}{P_{lh}} \quad (10)$$

where $n_{e20}(r_{max})$ is the electron density at r_{max} in units of 10^{20} m^{-3} , and I_{lh} and P_{lh} are the lower hybrid driven current and absorbed power, respectively. To compute η_{lh} we use the expression due to Fisch [10], multiplied by an adjustment factor to bring it into agreement with detailed calculations of ray tracing and power deposition [11]. This correction factor is typically 0.6.

For the NB figure of merit we employ an expression which can be obtained from an analytic expression for the fast ion distribution function [12]. The expressions reads

$$\eta_{nb} = \frac{\langle n_{e20} \rangle I_{nb} R_0}{P_{nb}} = \frac{C_{nb} \langle T_e \rangle J(\langle x \rangle, \langle y \rangle) \langle F_p \rangle}{\ln \Lambda} \quad (11)$$

where $\langle \rangle$ denotes a volume average and $J(x,y)$ is a function which takes into account the appropriate moment of the fast ion distribution function and is approximated by [13]

$$J(x,y) = x^2 / [4 + 3y + x^2(x + 1.39 + 0.69y^{0.7})] \quad (12)$$

$x^2 = E_p/E_c = E_p/10A_b T_e$, $y = 42_{eff}/5A_b$, A_b is the beam mass in amu, E_p is the NB energy, $\ln \Lambda$ is the Coulomb logarithm for electrons, and C_{nb} is a correction factor to fit the detailed calculations of beam deposition and fast ion current performed with the FPT code [14]. $\langle F_p \rangle$ is a factor to account for the partial cancellation of the fast-ion current by the electrons and is typically 0.76.

Steady State Operating Contours (SOPCON's)

In the case of ignition, Plasma OPERating CONtours, or POPCON's, have proven to be a convenient way of representing the operating space for the plasma. In POPCON's, contours of external heating power required to balance the losses are plotted on the $\langle n_e \rangle$ - $\langle T \rangle$ plane. The ignition boundary is then clearly shown where the external power goes to zero. Other limits on density and beta are easily seen. Such POPCON's are plotted for a given value of the plasma current and an assumed energy confinement scaling law.

POPCON's are not as convenient for examining operating space for steady state operation. We have found another type of plot to be

more useful, namely, contours of Q = fusion power/current-drive power, $\langle \Gamma \rangle$ = average neutron wall load and current-drive power, $P_{cd} = P_{th} + P_{nb}$, in the I_p - C_T plane. One such Steady state Operating Contour, or SOPCON, is shown in Fig. 1.

The power balances for electrons and ions are performed separately for each point on the SOPCON. Alpha power and current drive power are balanced against transport losses and radiation power. These equations determine the electron and ion temperatures. For a fixed configuration, each point in I_p - C_T space corresponds to a unique value of beta. This uniquely determines the plasma density at power balance. These equations and the other physics expressions described above are incorporated into the MUMAK code [15].

For the SOPCON shown in Fig. 1, we have adopted Kaye-Goldston scaling with $H = 1.3$, assumed $Z_{eff} = 2.0$ and that 5% of the alpha particle power is promptly lost. The major and minor radii are taken as 4 and 1.4 m respectively and the elongation and triangularity are 2.3 and 0.6 at the separatrix, respectively, corresponding to 2.0 and 0.4 at the 95% flux surface. We further assume that 30% of the current is available from neoclassical (bootstrap) effects. The NB energy was taken to be $E_0 = 1.5$ MeV to assure good penetration to the axis for current drive. The LH frequency was $f_0 = 8$ GHz to assure that parametric instabilities at the plasma edge and parasitic damping on fast alpha particles are avoided. The LH launched spectrum was assumed to be centered about $N_{||} = 1.7$ to achieve maximum penetration.

This plot clearly shows the boundaries furnished by the requirements described in the previous section. Contours for plasma Q values of 5.0 and 7.5, total current drive power equal to 70 and 100 MW, neutron wall load equal to 1.0 and 1.5 MW/m², and safety factor q , equal to 3 and 4 are shown. In addition, the shaded region shows the fraction of the operating space for which the power balance occurs with a density exceeding the Greenwald density limit.

Based on the SOPCON in Fig. 1, we have several possible choices for our operation scenario. We could choose to operate on the $\langle \Gamma \rangle = 1$ MW/m² contour. In this case we have a choice of operating at the highest possible $Q \approx 6.5$ which occurs at $q_0 \approx 3.0$, or take $Q = 5$ and 100 MW of current-drive power so as to operate at higher safety factor. Note that all of these conditions occur with a conservative Troyon coefficient of about 2.5%. Along this contour, the volume-averaged electron density ranges from 0.7 at high Q to 0.9×10^{20} m⁻³ at the lower Q . The electron (ion) temperature ranges from 21 (19) to 16 (14) keV.

If we wish to operate with higher wall loading, Fig. 1 tells us that we must operate at a higher Troyon coefficient; $\langle \Gamma \rangle \approx 1.5$ MW/m² occurs at $C_T \approx 2.9$. At this point, $Q = 7.5$ is attainable if we accept a q_0 of about 3.2, but if we wish a higher value of

safety factor, we will need more current-drive power leading to a lower Q value. It should be noted that we cannot actually control the Troyon coefficient, rather it is a convenient measure of the operational beta. In an experiment, it would be necessary to control the fueling rate and the current-drive power to achieve the desired beta.

In Fig. 2 we show a SOPCON for the same assumptions and systems parameters except that Z_{eff} is lowered to 1.5. The same general picture emerges, but the required current-drive power is somewhat less and the Q is somewhat higher for any point in the I_p - C_T plane.

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FIGURES

Figure 1. SOPCON for a 4 m ITER with $Z_{eff} = 2.0$.

Figure 2. SOPCON for a 4 m ITER with $Z_{eff} = 1.5$.

FIGURE 1.

at $n = 1.0$, alt. $\cdot 5$, $E_D = 1.50 \text{ MeV}$, $F_a = .95$, $Z_{eff} = 2.0$

FIGURE 2.

CHOOSING A STEADY STATE OPERATING POINT

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**Presented at the ITER Current Drive and Heating Preparatory Meeting
Princeton Plasma Physics Laboratory, Princeton NJ**

June 9, 1988

STEADY STATE OPERATION IS ATTRACTIVE EVEN WITH CONSERVATIVE PHYSICS

- Optimized steady state operation occurs with enhanced (diverted) L-mode confinement, $\tau_E = H \tau_{\text{Goldston}}$ with $H \approx 1.5$.
 - H-mode is essentially a transient phenomenon (ELMs, density rise etc.); L-mode is consistent with steady state.
 - Allows more peaked density profiles than H-mode
 - * Higher wall loading for the same average density and temperature
 - * Enhanced probability of bootstrap current
- Optimized steady state operation occurs well below the density and beta limits.
 - Reduced frequency of disruptions

CONSTRAINING RELATIONS

- Enhanced L-mode: $r_E = 1.5 r_{\text{Goldston}}$
- Troyon Beta Limit: Troyon coefficient, $C_T \leq 0.03$
- Safety Factor Limit: $q_p \geq 3.0 \Rightarrow \text{Current } I_p \leq I_p(\text{max})$
- Greenwald Density Limit: Leading coefficient, $f_{gr} = 1.0$
- Lower Hybrid Spectrum: $N_{||}^{\text{sc}} \leq N_{||} \leq N_{||}^{\text{ELD}}$
- Beam Penetration: $E_{nb} \propto L_b$ (path length)
- Bootstrap Current: $I_{bs} = 0.3 I_p$ (fixed for the calculations)

CASE PARAMETERS

R_o [m]	4.0	5.0	6.0
a [m]	1.41	1.75	2.10
I_p (max) [MA]	16.4	20.5	24.5
E_{nb} [MeV]	<u>1.5</u>	<u>1.86</u>	<u>2.23</u>

Constant: $\kappa = 2.3$, $\kappa_{95} = 2.0$, $\delta = 0.6$, $\delta_{95} = 0.4$

$B_o = 5.0$ T, $\Delta_{fw} = 0.11$ m, $f_{LH} = 8.0$ GHz

CONTOURS SHOWN ON THE SOPCONS

Average Neutron Wall Loading: $\langle \Gamma \rangle = 1.0, 1.5 \text{ MW/m}^2$

Plasma Power Ratio: $Q = 5, 7.5, 10, 15$

Total Current Drive Power: $P_{CD} = 70, 100 \text{ MW}$

Surface Heat Load: $W_{div} = 1.0, 1.2 \cdot (\text{U.S. ITER Baseline})$

Edge Safety Factor (estimate): $q_{\psi} = 3.0, 4.0$

Greenwald Density Limit (shaded): $n_e \geq 1.0 \cdot n_{Gr}$

"ITER" WITH $R_0 = 4\text{ m}$ $B_0 = 5\text{ T}$ $Z_{eff} = 1.5$
 $a = 1.41\text{ m}$ $T_E = 1.5\text{ T}_{coolston}$

$R_{DIV} = W_{DIV} / W_{DIV}(U.S. \text{ BASELINE})$

$81\text{N} \cdot 1.8$, $81\text{T} \cdot 1.8$, $81\text{H} \cdot 1.8$, $81\text{C} \cdot 1.8$, $81\text{D} \cdot 1.8$, $81\text{E} \cdot 1.8$, $81\text{F} \cdot 1.8$, $81\text{G} \cdot 1.8$, $81\text{H} \cdot 1.8$, $81\text{I} \cdot 1.8$, $81\text{J} \cdot 1.8$, $81\text{K} \cdot 1.8$, $81\text{L} \cdot 1.8$, $81\text{M} \cdot 1.8$, $81\text{N} \cdot 1.8$, $81\text{O} \cdot 1.8$, $81\text{P} \cdot 1.8$, $81\text{Q} \cdot 1.8$, $81\text{R} \cdot 1.8$, $81\text{S} \cdot 1.8$, $81\text{T} \cdot 1.8$, $81\text{U} \cdot 1.8$, $81\text{V} \cdot 1.8$, $81\text{W} \cdot 1.8$, $81\text{X} \cdot 1.8$, $81\text{Y} \cdot 1.8$, $81\text{Z} \cdot 1.8$

"ITER" WITH

三三三

三才圖會

$B_0 = 5T$

**○
○
○
○**

Q = 1.417m

$$T_E = 1.5 T_{coldston}$$
$$R_{110} = W_{110}/V_{110} \text{ (u.s. gallons/lb)}$$

"ITER" WITH $R_0 = 5m$ $B_0 = 5T$ $Z_{eff} = 1.5$
 $a = 1.75m$ $T_E = 1.5 T_{GOLDSTON}$
 $R_{DIV} = W_{DIV} / W_{DIV} (V.S. BASELINE)$

81m 1.0. alt. HETS, JEU081906MBV1.68e+0100, Zeff = 1.50

"ITER" WITH $R = 5m$ $\theta_0 = 5T$ $B_{eff} = 2.0$
 $a = 1.75m$ $T_E = 1.5$ GOLDSTON

$R_{DIV} = W_{DIV}/W_{DIV}$ (U.S. ONSELINE)

88. 5.00m. a. 1.75m. k. 2.0. d. 4. 80. 4.98T. H. 1.50

[illegible]

PLASMA CURRENT, I_p (kA)

Fig. 8.60a. $Q = 2.13$, $K = 2.5$, $\langle P \rangle = 1.0 \text{ MW/m}^2$

$R_{01V} = W_{01V}/W_{01V}(0.5, 0.95 \text{ eV})$

$T_e = 1.5 \text{ eV}$

$a = 2.1 \text{ m}$
 $M = 0.01$
 $W = 0.01$
 $W_{01V} = 0.01$

OPERATING POINTS - $Z_{eff} = 2.0$

R_o [m]	4.0	5.0	6.0
I_p (I_{bo}) [MA]	16 (4.8)	20 (6.0)	24 (7.2)
G_T (t)	2.3	2.55	2.7
T_e (T_i) [keV]	17.5 (15)	22 (20)	26 (26)
n_e (n_{gr}) [$10^{20} m^{-3}$]	0.88 (1.7)	0.71 (1.4)	0.57 (1.1)
$\langle T \rangle$ [MW/m ²]	1.03	1.23	1.37
Q	<u>5.2</u>	<u>9.4</u>	<u>13.7</u>
γ_{rb} (γ_{lh}) [$10^{20} A/Wm^2$]	0.42(0.35)	0.52(0.40)	0.61(0.44)
P_{rb} (P_{lh}) [MW]	69 (28)	78 (20)	82 (13)
P_{CD} [MW]	<u>97</u>	<u>98</u>	<u>95</u>
P_{fus} [MW]	504	920	1310

OPERATING POINTS - $Z_{eff} = 1.5$

R_o [m]	4.0	5.0	6.0
I_p (I_{bs}) [MA]	16 (4.8)	20 (6.0)	24 (7.2)
C_T (%)	2.3	2.5	2.8
T_e (T_i) [keV]	18.3 (16)	23 (21)	28 (29)
n_e (n_{or}) [$10^{20} m^{-3}$]	0.80 (1.7)	0.61 (1.4)	0.53 (1.2)
$\langle \Gamma \rangle$ [MW/m ²]	1.01	1.13	1.25
Q	<u>6.3</u>	<u>11.0</u>	<u>17.7</u>
γ_{nb} (γ_{lh}) [$10^{20} A/Wm^2$]	0.46(0.41)	0.57(0.47)	0.67(0.51)
P_{nb} (P_{lh}) [MW]	59 (20)	63 (12)	70 (20)
P_{CD} [MW]	<u>79</u>	<u>75</u>	<u>80</u>
P_{fus} [MW]	500	850	1330

CONCLUSIONS

- 0 Total current drive power requirement is nearly constant for devices with major radii, $R_0 = 4, 5, \text{ and } 6$ meters providing that Alfvén instabilities do not severely degrade NBI current drive efficiency.
- 0 Optimized steady state operation occurs with enhanced (diverted) L-mode energy confinement.
- 0 Steady state operation satisfying the ITER goals is possible with density well below the density limit, beta below the beta limit, and current below the maximum value (q limit).

MEMORANDUM

July 22, 1988
LLNL-ITER-88-049

To: ITER physics distribution

From: S.K. Ho and M.E. Fenstermacher

Subject: POPCON thermal stability analysis for steady state operation

Summary

We consider the thermal stability of steady state operating points for ITER using POPCON analysis (contours of the required auxiliary power for power balance in a plasma density - vs. - temperature space). The auxiliary power requirement at each (n, T) point is calculated from a power balance which includes alpha power, transport power losses, radiation power losses, and non-inductive current drive power. We find that for full H-mode energy confinement ($\tau_E = H\tau_{L-mode}$, $H = 2.0$), the current drive efficiencies are sufficiently low that the sum of the power required to sustain the current plus the alpha power exceeds the total plasma losses for all values of (n, T) below the beta limit. The plasma is not in thermal equilibrium under these conditions. For lower values of the H-mode enhancement factor, thermal equilibrium "windows" in (n, T) operating space exist. For "enhanced L-mode" confinement ($H = 1.5$), regions in the thermal equilibrium windows are identified which are stable to temperature excursions. A further reduction in H to the L-mode confinement ($H = 1.0$) results in a large external heating power requirement.

Analysis

Plasma OPERating CONtours (POPCONS) have been generated for steady state operation of ITER. The auxiliary power P_{aux} required for power balance at each point in (n, T) space is calculated from

$$P_\alpha + P_{cd} + P_{aux} - P_{tran} - P_{rad} = 0$$

where P_α , P_{cd} , P_{tran} , and P_{rad} are the alpha, current drive, transport loss, and radiation loss powers, respectively. The required current drive power is determined from

$$P_{cd} = \frac{I_{cd}}{\eta_{cd}} = \frac{I_p - I_{bs}}{\eta_{cd}}$$

where I_p is the total plasma current, I_{bs} is the bootstrap current which is assumed to scale as

$$I_{bs} \propto I_p \sqrt{\epsilon} \beta_p$$

with ϵ and β_p being the inverse aspect ratio and poloidal beta, and η_{cd} is the current drive conversion efficiency (A/W).

Note that in regions for which $P_\alpha + P_{cd}$ exceeds the total transport plus radiation losses, the auxiliary power calculated will be negative. This means that thermal equilibrium is not possible in these regions without an additional power loss mechanism. Thermal equilibrium exists only in regions which satisfy:

$$P_{aux} \geq 0$$

The thermal equilibrium is maintained either by current-drive power alone ($P_{aux} = 0$) or by the current drive power with additional auxiliary heating ($P_{aux} > 0$).

An unstable excursion to a higher temperature (T) is possible if the local external heating power $P_{cd} + P_{aux}$ at the higher T is less than that at the equilibrium. Hence, we can define the criterion for thermal stability at any operating point (n, T) as:

$$\frac{\partial P_{total}}{\partial T} \equiv \frac{\partial}{\partial T}(P_{aux} + P_{cd}) > 0$$

Constant density is assumed in this stability criterion because the particle confinement time is much longer than the energy confinement time. The thermal stability criterion is illustrated in Fig. 1 which shows, as an example, the auxiliary heating power contours for an inductive pulsed mode ($P_{cd} = 0$). The alpha ignition ($P_{aux} = 0$) is stable above 10 – 13 keV since $\partial P_{aux}/\partial T > 0$.

We will examine 3 cases of a 5.8 m ITER machine using Goldston L-mode energy confinement scaling with enhancement factor H of 2.0, 1.5, and 1.0. We use a combined neutral beam and lower hybrid current drive scheme for this study. Note that P_{cd} is only a function of the point in (n, T) space and not a function of H . Therefore $P_{aux}(n, T)$ will vary with H to satisfy power balance. Figure 2 shows P_{cd} in the (n, T) space for a fixed total plasma current I_p of 20 MA. The contours of P_{cd} reflect the effects of increasing η_{cd} with increasing T at constant density. At constant temperature, the shape of the contours shows the competition between the decrease in η_{cd} with increasing n (low density) and the decrease in $I_{cd} = I_p - I_{bs}$ as β_p and hence I_{bs} increases (high density).

Contours of P_{aux} for the above mentioned cases ($H = 2.0, 1.5$, and 1.0) are shown in Figs. 3-5, respectively. The beta limit for a Troyon coefficient of 3% is also labeled. We observe that for $H = 2.0$, equilibrium occurs only for $T > 27$ keV because the current-drive power is excessive in a relatively low T and well confined plasma. These equilibria are well above the beta limit. The equilibrium windows in (n, T) space open up when H decreases. The equilibrium threshold temperature below the beta limit for $H = 1.5$ is about 22 keV at $n = 7 \times 10^{19} \text{ m}^{-3}$. For $H = 1.0$, the threshold is at $T = 8$ keV at $n = 2.3 \times 10^{20} \text{ m}^{-3}$. Figures 6 and 7 show the total external heating power $P_{cd} + P_{aux}$ for $H = 1.5$ and 1.0 , from which the thermally stable regions can be identified. We find that the regions of thermal equilibrium ($P_{aux} \geq 0$) with $H = 1.5$ are thermally stable. Those with $H = 1.0$ are only marginally stable as indicated by the relatively flat gradients of $P_{cd} - P_{aux}$ with respect to T .

For a neutron wall loading between 1 and 2 MW/m², we locate stable thermal equilibria for $H = 1.5$ at $T \sim 23 - 25$ keV and $n \sim 4 - 7 \times 10^{19}$ m⁻³ that require P_{cd} of 70 - 100 MW without P_{aux} , resulting in $Q \approx 10$. Similar results for $H = 1.0$ have $T \sim 13 - 16$ keV and $n \sim 6 - 9 \times 10^{19}$ m⁻³, $P_{cd} \sim 150 - 250$ MW, and $Q < 5$. The operation windows bounded by the wall loading, the thermally stable equilibrium contour ($P_{aux} = 0, \partial P_{total} / \partial T > 0$), and the beta limits for these 2 cases are shown in Fig. 8. The L-mode energy confinement has larger operating windows, but since the confinement is poor, undesirably large external heating power is required.

In summary, the POPCON thermal equilibrium and stability analysis provides an approach for selecting a steady state design point in (n, T) space for fixed machine parameters R, a, I_p, B_0 , etc. based on thermal stability considerations. For the example of the 5.8 m machine, a high confinement ($H = 2.0$) does not have thermal equilibrium operating points below the beta limit while a low confinement ($H \approx 1.0$) requires excessive P_{cd} and/or P_{aux} for power balance. A moderate enhancement over L-mode, $H = 1.5$, results in stable thermal equilibria and acceptable plasma performance with reasonable input power requirement.

Fig. 1. $P_{aux}(n, T)$ vs n and T for pulsed mode ($P_{id} = 0$).

Fig. 2 P_{cd} (MW) vs n and T

Fig 4. $P_{aux}(MW)$ vs n and T for $H=1.5$.

HGTS 0. 10 2.02e+02

5

Fig 5 P_{aux} (MW) vs n and T for $H = 1.0$.

N-Weighted Volume Averaged Temp., T_e, T_i (keV)

Fig 6 $P_{cd} + P_{aux}$ (MW) vs ρ and T for $H = 1.5$

Volume Averaged Density, N_e ($10^{20} m^{-3}$)

HGIS B.

10.2×10^{22}

Volume Averaged Density, n_e ($10^{20} m^{-3}$)

HGTS 0. 10 2.00e+02

N-Weighted Volume Averaged Temp., $T_e \cdot T_i$ (keV)Fig 7 $P_{ed} + P_{aux}$ (MW) vs n and T for $H = 1.0$

N - weighted volume averaged temperature, $T_e = T_i$ (keV)

Fig. 8 Operation windows bounded by the beta, wall loading (P), and thermal equilibrium and stability ($P_{max}=0$) limits

IV. Tokamak Specifications

We examine a range of tokamak parameters and look at the costs and benefits of size variations. Field and current levels are set by insistence on relevant values of β and q . The critical physics parameters are the scaling of energy confinement and current drive efficiency. Sufficient power density at the plasma surface is also a requirement. An important consideration is flexibility, to allow variation of plasma shape, and to permit possible upgrades in field, current, and power subject to cost effectiveness.

Magnetic Field and Plasma Current

For the STE experiments, the target plasma parameters are temperatures of about 10 keV and density around $1 \times 10^{20} \text{ m}^{-3}$. These parameters can be related to the machine parameters of size, field strength, and plasma current, by insisting that the STE tokamak operate in a plasma regime that is relevant to reactors.

The two key plasma parameters are β , which measures the plasma pressure in terms of the pressure of the external magnetic field, and the safety factor q , which constrains the plasma current. Values of β that are too high, or of q that are too low, lead to deteriorating energy confinement, reduced stability, and eventually disruption. On the other hand, high pressure (high nuclear power density in a reactor) and efficient use of the external field requires operation at high β . High β limits and good energy confinement are strongly correlated to large plasma current, which leads to operation at low q .

A number of stability analyses have found, and experiments tend to confirm, that the maximum stable value of β is proportional to the normalized plasma current, $I_N \equiv \mu_0 I / aB$, for a wide variety of conditions. We define the normalized value, $\beta_N \equiv 100 \times \beta / I_N$ (Troyon obtained the limit of $\beta_N \simeq 3$ for ballooning modes.) The STE device should be capable of operation close to these limits at its highest current and field capacity. If the center of the operating parameter space is near $\beta_N \sim 2$, this condition will be satisfied.

The limitation on q is also well defined. Experiment and theory both indicate that there is a hard stability limit that sets the minimum value of q , and energy confinement studies indicate that there is a deterioration in confinement as q is lowered toward this limit. For elongated plasmas ($\kappa \gtrsim 2$), operation with $q < 3$ appears to be difficult, and may have deleterious effects on stability and confinement.

The actual safety factor can be represented by the approximate "cylindrical q ",

$q_{cyl} \equiv (2\pi a^2 B / \mu_0 I R) \cdot (1 + \kappa^2)/2$, for parametric scaling studies. In defining the STE parameters we take $q_{cyl} \sim 2$. For the strongly shaped, diverted plasmas expected in the STE device, $q_{cyl} \simeq 2$ will correspond to a value of the safety factor at the 95% flux surface of $q(95) \simeq 2.8$ –3.5.

With fixed pressure, normalized β , and safety factor, the required toroidal magnetic field strength and plasma current become functions of plasma dimensions alone:

$$B_T \propto \left[\frac{\langle p \rangle q_{cyl}}{\beta_N \left(\frac{1+\kappa^2}{2} \right)} \right]^{1/2} \frac{R^{1/2}}{a^{1/2}}$$

$$I_P \propto \left[\frac{\langle p \rangle \left(\frac{1+\kappa^2}{2} \right)}{q_{cyl} \beta_N} \right]^{1/2} \frac{a^{3/2}}{R^{1/2}}$$

Size and Shaping

We have examined a range of devices consistent with these parameters. The fixed values are $\langle p \rangle = 0.3$ MPa, $q_{cyl} = 2.2$, $\beta_N = 1.7$ (corresponding to $\langle \beta_{95} \rangle = 2.2 I/aB$), and $\kappa = 2$. Where needed, the triangularity is taken to be $\delta = 0.4$. With this selection of parameters, the toroidal magnetic field strength at the minor axis is in the neighborhood of 5 tesla, and the plasma current is 3–5 MA. Specifically, the toroidal magnetic field is taken to be $B_T = 2.5 (R/a)^{1/2}$, and the plasma current is $I_P = 14 a^{3/2} / R^{1/2}$.

The independent variables are size, represented by the minor radius a , and the aspect ratio R/a . The cases considered range from $a = 0.3$ m to $a = 0.7$ m, and from $R/a = 2$ to $R/a = 6$. The major radius ranges from 1 m to 3 m. These values encompass the range of sizes and aspect ratios that are considered for the STE device. The smaller devices have the advantage of lower cost and lower power requirements. On the other hand, the smaller devices do not allow personnel access to the chamber for maintenance and repair. The plasma characteristics of the smaller devices also tends to be dominated by surface effects.

This parameter range does not represent a major extrapolation from the present experimental data base. Thus, empirical scaling rules based on present data should give reliable predictions of STE performance.

The requirement of personnel access sets the minimum height of the inside of the vessel at roughly 2 m. The fraction of the plasma dominated by surface effects is

approximately $2\lambda/a$, where λ is the mean free path of a neutral atom at the plasma edge. Setting $\lambda/a \leq 0.1$ gives a lower limit on the product of average plasma density and radius of $\langle n \rangle a \geq 0.5$.

High elongation and indentation have been shown to be beneficial for increasing the equilibrium β -limit or for access to the second-stability regime. In addition, possible improvements in plasma confinement have been suggested. There are several tokamaks now in operation or under construction (DIII-D, PBX, TCA) which have shape optimization studies as a major feature of their experimental program, and we expect that there will be further indication of the benefits and limitations of plasma shaping in the near future.

Stable operation of shaped plasmas depends on the ability to establish and control appropriate plasma current profiles. Generally, the current profile must be broader than attainable in steady-state without local current drive. This capability is a primary feature of the STE program, and it will allow investigation of a broader range of plasma shapes than is possible at present.

For the purpose here of determining the dependence of heating and current drive power, and magnet size on the basic plasma dimensions, we have fixed the elongation at $\kappa = 2$. The reference plasma shape is bounded by a separatrix that provides a double-null divertor.

Heating and Current Drive Power

The auxiliary power required to produce plasmas at temperatures and densities which are appropriate to the goals of STE has been calculated using the MUMAK volume averaged power balance code.¹ Here, we briefly describe the models and assumptions used, and present the results of comparisons for various energy confinement scaling assumptions and neutral beam injection energies, as well as aspect ratio and minor radius.

MUMAK is a zero-dimensional, time-independent power balance code for modeling particle and energy flow in tokamaks. The separate electron and ion equations include terms for noninductive current drive power, heating power, conduction power losses described by empirical energy confinement scaling laws, and radiation power losses. The third equation is a pressure balance constraint. These relations yield the average

¹ M. E. Fenstermacher, *MUMAK - A Computer Code for Modeling Plasma Power Balance and Current Drive in Tokamaks*, LLNL Report UCID-21038, April 1987.

electron density $n_e = \langle n_e(r) \rangle$, and the average electron and ion temperatures (density-weighted: e.g., $T_e = \langle n_e T_e \rangle / \langle n_e \rangle$). Fixed radial profiles are assumed (of the form $n(r) = n(0)[1 - (r/a)^2]^{\alpha_n}$, with $\alpha_n = 0.5$ and $\alpha_T = 1.0$). For these comparisons, we fix both the density and temperature, corresponding to $\langle p \rangle = 0.3$ MPa: $n_e = 1.0 \times 10^{20} \text{ m}^{-3}$ and $T_e = 9.35 \text{ keV}$ ($\langle T_e \rangle = 7.80 \text{ keV}$). Table 4.1 gives the results of these calculations.

In the power balance relations, the conduction power is calculated from an energy confinement time ($P_{i,e}^{\text{cond}} = W_{i,e} / \tau_{Ei,e}$, where $W_{i,e}$ is the total energy in the electron or ion fluid). The energy confinement scaling is assumed to follow "saturated H-mode" form:

$$\tau_E = \left[\frac{1}{\tau_{NA}^2} + \frac{1}{(2\tau_{KG})^2} \right]^{-1/2}$$

where the "neoAlcator" confinement time is

$$\tau_{NA} = 0.0625 \left(\frac{\bar{n}}{10^{20}} \right) a R^2 q \text{ sec}$$

and the "Kaye-Goldston" time is

$$\tau_{KG} = 0.055 \frac{\left(\frac{I}{10^4} \right)^{1.24} \left(\frac{n}{10^{20}} \right)^{0.26} \kappa^{0.25} R^{1.65} \left(\frac{M}{\sqrt{2}} \right)^{0.5}}{B^{0.09} a^{0.49} \left(\frac{P}{10^4} \right)^{0.36}} \text{ sec}$$

The safety factor used here is $q_0 \equiv (2\pi a^2 B / \mu_0 I R) \cdot [1 + \kappa^2(1 + 2\delta^2)]/2 = 2.80$. The confinement times for electrons and ions are assumed equal: $\tau_{Ei} = \tau_{Ee}$.

MUMAK also calculates noninductive current drive efficiencies for electron cyclotron, neutral beam, lower hybrid slow wave, and fast wave current drive, and calculates the bootstrap current as well. For these comparisons we have assumed that neutral beam injection (NBI) and lower hybrid (LH) waves will be used for current drive. Additional electron cyclotron (EC) heating power is used to achieve the desired bulk electron temperature. The NBI model includes the effects of electron trapping as well as the beam energy dependence. The LH models take into account both the accessibility limit and the condition for avoiding strong electron Landau damping of the wave to determine the radius of maximum penetration for the waves.

With fixed plasma density, temperature, and profiles, the current drive efficiencies do not vary over range of devices examined. The calculated NBI current drive efficiency is $\eta_{NBCD} = 0.25$ ($\eta = nIR/P$), and for lower hybrid waves it is $\eta_{LHCD} = 0.35$. The high NBCD efficiency is obtained with the use of 500 keV neutral beams, which

Table 4.1 - MUMAK Cases for STE

case #	1	2	3	4	5	6	7	8	9
R/a	2	3	3	3	4	4	4	4	4
R (m)	1.0	1.2	1.5	1.8	1.2	1.6	2.0	2.4	2.8
a (m)	0.5	0.4	0.5	0.6	0.3	0.4	0.5	0.6	0.7
B (T)	3.54	4.33	4.33	4.33	5.00	5.00	5.00	5.00	5.00
I_{b_0}	1.19	0.96	1.19	1.43	0.71	0.95	1.19	1.43	1.67
I_{NB}	1.22	0.70	0.94	1.19	0.37	0.57	0.78	0.99	1.20
I_{LH}	2.54	1.58	1.91	2.23	1.02	1.28	1.53	1.78	2.03
I (MA)	4.95	3.23	4.04	4.85	2.10	2.80	3.50	4.20	4.70
P_{NB}	6.6	4.6	7.7	11.0	2.6	4.7	7.3	10.4	14.2
f_{shine}	0.24	0.26	0.25	0.20	0.32	0.22	0.15	0.10	0.07
P_{LH}	7.8	5.5	8.4	11.8	3.5	6.0	8.9	12.3	16.4
P_{EC}	16.5	12.0	10.0	8.0	9.0	7.9	6.5	5.1	2.5
P_{IC}	22.5	15.8	16.8	18.0	12.5	13.2	14.4	15.9	17.5
P_{total} (MW)	53.4	37.9	42.9	48.8	27.6	31.7	37.1	43.7	50.6
τ_E (msec)	74	82	143	219	63	131	221	327	450
P/A (MW/m ²)	1.71	1.26	0.92	0.72	1.23	0.79	0.59	0.49	0.41

case #	10	11	12	13	A	B	C
R/a	5	5	5	6	3.64	2.57	3.18
R (m)	2.0	2.5	3.0	3.0	1.2	1.67	1.75
a (m)	0.4	0.5	0.6	0.5	0.33	0.65	0.55
B (T)	5.59	5.59	5.59	6.12	4.17	4.01	4.46
I_{b_0}	0.95	1.19	1.43	1.19	0.78	1.55	1.31
I_{NB}	0.49	0.67	0.85	0.59	0.46	1.46	1.03
I_{LH}	1.05	1.28	1.49	1.08	1.18	2.67	1.98
I (MA)	2.50	3.13	3.76	2.86	2.42	5.68	4.32
P_{NB}	4.8	7.6	11.1	7.8	3.1	11.0	8.4
f_{shine}	0.19	0.12	0.09	0.10	0.30	0.13	0.15
P_{LH}	6.1	9.2	12.9	9.2	4.1	13.3	10.2
P_{EC}	6.0	4.0	2.0	3.5	9.7	8.6	7.5
P_{IC}	12.0	13.5	15.0	12.4	13.4	21.5	18.0
P_{total} (MW)	28.9	34.3	41.0	32.8	30.3	54.4	44.0
τ_E (msec)	183	301	440	382	69	209	195
P/A (MW/m ²)	0.53	0.44	0.36	0.35	1.23	0.80	0.73

penetrate well (the shine-through fraction ranges from about 0.25 for the smaller devices to 0.1 for the larger ones). The LHCD efficiency is high because in these examples it is used for driving current near the plasma edge, and penetration is not a problem. Combining these two methods gives a net efficiency of 0.31.

Even though the selected profiles are not optimized for the bootstrap current, 20–40% of the total plasma current is bootstrap. For these examples, we find that $I_{boot} = 2.38 \text{ a MA}$, giving $I_{boot}/I = 0.17\sqrt{R/a}$. With the profile control expected in the STE device, significant improvement in I_{boot}/I is likely – reducing the power required for current drive.

The total power required to heat and drive current in the range of sample devices is shown in Fig. 4.1. This range straddles the transition between confinement dominated by τ_{NA} and by τ_{KG} . The range of power levels required is from below 30 MW for small plasmas at large aspect ratio to above 50 MW. At low aspect ratio and small minor radius, the power requirement is dominated by τ_{NA} . As either a or R/a is increased, the τ_{KG} scaling becomes more important. It is of interest to note the decrease in required heating power with increasing aspect ratio. This is a feature of all of the empirical L-mode confinement scaling rules.

The required current drive power is shown in Fig. 4.2. The power needed is roughly given by $P_{CD} = 45 a^2 \sqrt{R/a} (1 - 0.17\sqrt{R/a})$, where the second term results from the reduction of the external current drive requirement due to the increasing bootstrap fraction. In the range of interest, the dependence of current-drive power on aspect ratio is very weak. Note also that, for the estimated current drive efficiency, the current drive power is less than the required heating power throughout the size range. If either the efficiency is a factor of two lower, or energy confinement is a factor of two better, then the two power levels become comparable. In each case, the difference between required heating power and current drive power is assumed to be supplied by a combination of ECH and ICH.

In Fig. 4.3, we plot the average surface power density. This parameter, the heating power from Fig. 4.1 divided by the plasma surface area is indicative of the capability of the STE device for testing high heat flux plasma interactive components. The required range, heat flux exceeding 0.3 MW/m^2 , is met throughout the parameter space considered, but very high values are quickly reached for small a and small R/a .

In order to provide an indication of the range of performance capabilities of these devices, the operating space of the version with $a = 0.5 \text{ m}$ and $R/a = 4$ ($B = 5 \text{ T}$ and $I = 3.5 \text{ MA}$) is shown in Fig. 4.4. The axes are average density and temperature

Fig. 4.1 - Contours of total power (in MW) required as a function of minor radius and aspect ratio. For the cases considered, this is the heating power; the current drive power is smaller for all cases. The weak dependence on minor radius at low aspect ratio results from the neoAlcator portion of the τ_E scaling rule.

Fig. 4.2 - Contours of power required for current drive (in MW). Note the weak dependence on aspect ratio.

Fig. 4.3 - Contours of surface power density as a function of geometry. This is estimated by dividing the total power by the approximate plasma surface area.

(not density weighted), and the principal contours show the external heating power required at any $\langle n \rangle$ and $\langle T \rangle$. The pairs of small arrows indicate the Murakami and Greenwald density limits. The three hyperbolas represent constant β curves at $\beta_N = 1, 2$, and 3 . The dot-dash curves labelled $0.1-0.5$ indicate the minimum current drive efficiency needed to drive the plasma current at any set of parameters. At the reference operating point, $\eta \simeq 0.2$ would be sufficient.

Coil Specifications

In addition to these physics performance analyses, we have looked at the engineering characteristics of this range of devices. The key feature is the use of superconducting coils in both the toroidal and poloidal field systems. The analysis shows a clear distinction between "compact" ($R/a \lesssim 3$) devices and those with somewhat larger aspect ratio. It should be noted that the exact numerical results of this analysis depend on a number of arbitrary (but not unreasonable) assumptions.

The first important conclusion from this survey is that there are two different characteristic systems, depending on the performance of the poloidal field system. The smaller aspect ratio devices have little space inboard of the TF coils for the PF solenoid. This is illustrated in Fig. 4.5, which shows contours of the inductive capability of the solenoid. This quantity, $\tau = (\Psi - 2LI)/V_{loop}$, is a measure of the capability of the PF system to provide inductive flux in excess of that needed to initiate the plasma current. To the left of the curve $\tau = 0$ there is insufficient room for a solenoid that can bring the plasma current to its design value. For slightly larger devices, the inductive capacity of the solenoid increases rapidly.

The overall size of the coil system (and roughly its cost) is gauged by the coil mass. In Fig. 4.6 we plot the coil mass (toroidal and poloidal field systems) against aspect ratio for various minor radii. For $R/a > 3$, the mass is very closely proportional to R^2 . For $R < 3$, the mass decreases more rapidly. With decreasing size, first the PF solenoid gets squeezed out, and finally there is a limit below which it is not possible to specify a TF coil.

Observations

In these analyses we have avoided direct estimates of cost. Instead we have evaluated the two major elements that enter into a cost calculation, the magnet mass and the heating power requirement. The conversion to cost depends on the conductor and type of construction assumed for the magnets, and on the frequency range of the rf systems and energy of the neutral beams.

Date: February 2, 1988

To: ITER Physics Distribution

From: M. E. Fenstermacher and R. S. Devoto

Subject: Estimate of the Auxiliary Power Required to Ignite ITER

SUMMARY

We estimate the required heating power to make the transition from an Ohmic equilibrium to ignition to be approximately 30 MW in the US ITER baseline. This power could be supplied by lower hybrid heating (LHH), electron cyclotron resonance heating (ECRH), by ion cyclotron resonance heating (ICRH), or by neutral beam injection (NBI). From a comparison of these methods, we conclude that ECRH and NBI are the preferred heating methods.

MOTIVATION

The auxiliary power systems for ITER must serve two functions. Under pulsed operation they must provide sufficient heating to reach ignited burn in a reasonable time. In steady state operation both non-inductive current drive and plasma heating are required. On the basis of the present scientific data base, a combination of neutral beam injection and lower hybrid slow wave power appears to be the preferred method for steady state operation. In this memo we estimate the power required to heat ITER to ignition in pulsed operation. This represents an expansion of work begun by Logan. We consider the possible applicability of the steady state systems for heating to ignition as well as the possible use of ECRH or ICRH waves.

ESTIMATE OF THE REQUIRED POWER

We have estimated the required heating power to achieve ignition using the nominal parameters for the ITER pulsed scenario²: $R_0 = 4.04$ m, $a = 1.41$ m, $\kappa_0 = 2.3$, $\delta_0 = 0.6$, $B_0 = 4.982$ T, $I_p = 16.4$ MA, density exponent $\alpha_n = 0.5$, temperature exponent $\alpha_T = 1.0$, alpha concentration $C_\alpha = 5.6\%$ and $Z_{eff} = 1.5$. POPCON plots are shown in Figs 1 and 2 for "H-mode" and "L-mode" confinement respectively. In each case the global energy confinement time, τ_E , is given by

$$\tau_E = [r_{MA}^{-2} + \langle H r_{KE} \rangle^{-2}]^{-1/2} \quad (1)$$

where $H = 2$ is used to represent "H-mode" and $H = 1$ for "L-mode". Expressions for the Neo-Alcator time τ_{NA} and the Kaye-Goldston time τ_{KG} are given in the Appendix.

We will refer to several features of the POPCON's in our discussions. The "Ohmic equilibrium" contour is the zero auxiliary power curve at low temperature (see Fig. 1). On this curve the Ohmic power is the dominant heating power. The "ignition contour" is the zero auxiliary power curve at high temperature, where alpha power is sufficient to sustain the plasma. The "Cordey pass" is the saddle point region in which both $\partial P_{aux}/\partial T$ and $\partial P_{aux}/\partial n$ change sign.

The transition from the Ohmic operation to ignition follows by application of sufficient power to make both the transition to H-mode and then follow a trajectory to the ignition region where the alpha power is adequate to sustain the plasma. Comparison of Figs. 1 and 2 shows that the Ohmic equilibrium contour is the same for both L and H-mode, indicating that the energy confinement is dominated by the Neo-Alcator (Ohmic) term in Eq.(1). It should be possible to establish an Ohmic plasma at the Murakami density limit before application of sufficient auxiliary power to make the transition into H-mode and then to ignition.

Following Logan, we estimate the power required to make the transition from the Ohmic contour through the "Cordey pass" to ignition from

$$P_{aux} = P_{pass} + (W_{ign} - W_{ohm})/\tau_{heat} \quad (2)$$

where the stored energy is given by

$$W_i = 0.048 \langle n_{e20} \rangle_i (\langle nT_e \rangle / \langle n \rangle)_i V \quad (MJ) \quad (3)$$

$\langle n_{e20} \rangle$ (m^{-3}) and $\langle nT_e \rangle / \langle n \rangle$ (keV) are the ordinate and abscissa of the POPCON, respectively, and $V = 345 m^3$ is the plasma volume for ITER. Here W_{ign} is the plasma stored energy on the ignition curve (Point "A") and W_{ohm} is the stored energy on the ohmic equilibrium curve (Point "B"). From Fig. 1 we obtain $W_{ign} = 150 MJ$ and $W_{ohm} = 40 MJ$. Of course, the minimum required auxiliary power is just the power in the Cordey pass, but to achieve ignition with power approaching this level requires very long heating times. To keep a reasonable level of volt-sec consumption during heating, we choose $\tau_{heat} = 5 s$. The loop voltage at Point B is $V_l = 0.6 V$ so the heating phase would consume less than 3 V-sec out of a total of 108 V-sec available from the Ohmic transformer. The Ohmic energy confinement time at this starting point is also approximately 5s. From the Cordey pass power of 9.7 MW and Eq(2) we obtain the total required auxiliary power of approximately $P_{aux} = 30 MW$.

DISCUSSION

This estimate takes into account only the conduction losses calculated consistent with the auxiliary power shown on the POPCON's. Since the Kaye-Goldston confinement time degrades with auxiliary power, application of additional heating power to push through the Cordey pass will reduce the effective confinement time of the plasma. Table 1 shows the effect of adding 30 MW of auxiliary power to the power balance at Points "A" and "B" and at the Cordey pass, Point "C". We see that the chosen heating time is about the same as the energy confinement time at B with no auxiliary power and a little less than twice the energy confinement time at point A in the ignition regime. Since the energy confinement time decreases when we apply the 30 MW of power, the actual time to reach ignition will be somewhat longer than the ideal $\tau_{\text{heat}} = 5$ s.

Table 1. Energy confinement times with and without start-up power.

	Point B		Point C		Point A	
$\langle nT_e \rangle / \langle n \rangle$ (keV)	2.44		5.25		10.0	
$\langle n_{e20} \rangle$ (m^{-3})	0.91		0.91		0.91	
I_{fus} (MW)	3.4		51.		260.	
P_{aux} (MW)	0.	30.	9.7	40.	0.	30.
τ_{NA} (s)	4.9	4.9	4.9	4.9	4.9	4.9
τ_{KG} (s)	>50	5.8	9.2	4.4	4.8	3.3
τ_{E} (s)	4.9	3.8	4.3	3.3	3.4	2.7

The trajectory we have chosen is not unique. If we assume, however, that the starting point must be below the Murakami density limit (Ohmic equilibrium) and that the transit time to limit volt-sec consumption is $\tau_{\text{heat}} = 5$ s, then a trajectory from Point B to the point on the ignition curve with the minimum stored energy will give the minimum power requirement. Alternatively, the minimum stored energy point ("A'") could be reached in a somewhat shorter time with $P_{\text{aux}} = 30$ MW than from point B to A.

The results above are sensitive to the effective charge of the plasma. For comparison, we show the POPCON for ITER with $Z_{\text{eff}} = 2.0$ in Fig. 3. The ideal time required to reach ignition (Point A') with $P_{\text{aux}} = 30$ MW is $\tau_{\text{heat}} = 7.8$ s. Note that the density must exceed the Murakami density limit to reach ignition at $Z_{\text{eff}} = 2.0$. Since this density limit has been exceeded in a number of experiments with pellet fueling and/or

auxiliary heating, ignition should still be possible. The highest attainable density under these conditions is probably better represented by the Greenwald limit also shown in Figs. 1-3.

COMPARISON OF HEATING METHODS

Several heating methods are available for supplying the power to reach ignition: rf at lower hybrid (LH), ion cyclotron (IC), or electron cyclotron (EC) frequencies or neutral-beam injection (NBI). Lower hybrid waves at $f = 4.6$ GHz are proposed for driving edge currents in steady state operation and also for broadening the current profile in the pulsed mode. Some 20 MW of power is needed for steady state current drive but could be increased for ignition heating. However, experiments have shown that there is a density limit beyond which LH waves do not penetrate, apparently because of the excitation of parametric decay instabilities on the plasma edge. This limit is proportional to f^2 and appears to be slightly less than a line density of $2 \times 10^{20} \text{ m}^{-3}$ for $f = 4.6$ GHz (Alcator C data). In the Frascati Tokamak Upgrade (FTU), a frequency of 8 GHz was chosen to attempt to heat at line-average densities of $4\text{-}5 \times 10^{20} \text{ m}^{-3}$. Experimental data should be available this year from FTU to confirm that it is possible to heat with LH at higher densities.

Ion cyclotron heating of a minority ion is being used successfully in JET, but suffers from several flaws. It is, first of all, much more difficult to attain H-mode with IC heating. After H-mode is attained, say by the use of neutral beams, the changed edge conditions make it more difficult to couple the IC waves to the plasma. Another, perhaps fatal, defect observed in JET and ASDEX is the large amount of plasma impurities which apparently come from the antenna or Faraday shield under high-power operation. The latter problem might disappear if wave guides could be used for delivering the power, but this technique has yet to be demonstrated.

Both of the remaining heating methods are good candidates for heating to ignition. We presently predict the need for from 50 to 90 MW of 1.6 MeV NBI for driving current in the core region under steady state conditions. The higher power would be needed if fast-ion-induced Alfvén instabilities³ substantially degrade the current-drive efficiency, while the lower value is applicable if there is no degradation of the classical predictions. A beam energy of about 1.6 MeV is required to obtain approximately a 2 e-fold attenuation to the plasma center for beams aimed tangential to the geometric major radius of 4 m with a line-average plasma density of 10^{20} m^{-3} . Excitation-enhanced beam stopping cross sections are used for this estimate. If the ignition plasma is densified to the Murakami line-average density limit of about 10^{20} m^{-3} before heating, then the full-energy beams could be used. The current

beam injector design would allow operation at half energy, so for the same attenuation length, 25-45 MW of beam power would be available to heat a plasma of $0.5 \times 10^{20} \text{ m}^{-3}$ line-average density in the initial phase of heating. The full energy could be turned on during the plasma densification which results from the transition to H-mode and/or increased fueling near the axis.

A disadvantage with NBI heating is the non-localized power deposition. We would prefer to be able to heat in a small region near the axis so as to peak the temperature profile where the density is highest and thus facilitate ignition. Because the neutral beam deposits fast ions along its entire path in the plasma, the heating is more diffuse. Theory indicates that EC heating could be made quite localized near the resonance surface which we could place at the plasma center. For a toroidal field of 5 T, the resonance frequency is 140 GHz and the cut-off density is $2.4 \times 10^{20} \text{ m}^{-3}$. Thus, we could use fundamental O-mode launched from the outside in the baseline pulsed scenario with a peak density of about $2.2 \times 10^{20} \text{ m}^{-3}$. Since it is desirable to be able to heat at still higher densities, perhaps even above the Greenwald limit (see Fig. 1), it seems preferable to use 2nd harmonic X- or O-mode from the outside at 280 GHz. Density limits for these cases are about 4.8 and $9.7 \times 10^{20} \text{ m}^{-3}$, respectively. Absorption should still be adequate at this frequency and peaked near the axis although there will probably be a small parasitic absorption off axis from the next harmonic for rays which are not perfectly normal to the magnetic field. Arnold Kritz has verified this picture with calculations using linear absorption at conditions corresponding approximately to point A in Fig. 1. Further calculations are needed for point B. In addition, one needs to check for non-linear effects in the absorption. However, at this point, a free-electron laser system furnishing 30 MW appears to be a good candidate for heating to ignition in ITER.

CONCLUSIONS

We estimate the required heating power for ignition in ITER to be 30 MW. Heating from an ohmic equilibrium to ignition would be done in approximately 5 s, consuming less than 3 V-s. These results are sensitive to plasma effective charge; much longer heating times or increased power would be necessary if the plasma charge increases substantially from $Z_{\text{eff}} = 1.5$. The preferred heating methods to achieve ignition are ECRH and NBI because of their good penetration and freedom from major defects. Alternative scenarios could use a combination of heating methods. For example, one could use a lower ECRH power, say 10 MW, together with NBI and/or LH for heating should the cost of a 30 MW ECRH system be excessive. At least 10 MW of ECRH is desirable to aid in controlling instabilities, both in the pulsed operation and for steady state.

APPENDIX

The POPCON's in Figs. 1-3 were generated using the MUMAK power balance code.^{A1} Several refinements of the code calculations have been made since it was used to generate the results in Ref. 1. These changes were made to more closely align our calculation for ITER with the assumptions used in ignition calculations for CIT. The Neo-Alcator confinement scaling formula now reflects the latest CIT evaluation including TFTR data.^{A2} We have used

$$\tau_{MA} = 0.0625 \bar{n}_{e20} a R_0^2 q_{95x}$$

where $\bar{n}_{e20}$ is the line averaged electron density, and q_{95x} has been estimated for ITER in pulsed operation as $q_{95x} = 3.2$. For reference, some of the suggested scaling formulas for q give

$$q_0 = 5 a^2 B_0 / R_0 I_p = 0.75$$

$$q^* = q_0 [\int dl / 2\pi a]^2 = 2.30$$

$$q_{\kappa} = q_0 (1 + \kappa^2) / 2 = 2.36$$

$$q_f = q_0 [1 + \kappa^2(1 + 2\delta^2)] / 2 = 3.78$$

The Kaye-Goldston scaling formula has also been made consistent with CIT usage

$$\tau_{KG} = 0.055 I_p^{1.24} P^{-.56} R^{1.65} a^{-.49} \kappa^{.28} \bar{n}_{e20}^{-.26} B^{-.09} A_i^{0.5} \text{ (s)}$$

where $P = P_{\alpha} + P_{aux} - P_{rad}$ and $A_i = (A_D + A_T) / (A_D + A_T) = 1.67$.

The safety factor dependence of the "Murakami" density limit scaling has also been made consistent with CIT.^{A3} We have used

$$\bar{n}_{Mu} = 1.5 B_0 / R_a q_{eng}$$

where $q_{eng} = q_0 \kappa = 1.73$ for ITER. The "Greenwald" density limit marked on the POPCONs is given by

$$\bar{n}_{Gr} = 0.75 \kappa I_p / (\pi a^2).$$

Both the Murakami and Greenwald values are taken to be limits on the line-averaged density. For ITER profiles, the line-to-volume averaged density ratio is $\bar{n} / \langle n \rangle = 1.18$.

Finally, a report by Tani^{A4} on the variation of the ripple loss of fast alphas with q_i indicates that at the projected ITER value of $q_i \approx 3.2$ we can expect to lose approximately 10% of the alpha power. This loss has also been included in our calculations. It should be noted that Tani finds a very strong

increase in the ripple loss of alphas with increasing q_0 . Thus, we should operate at the lowest possible q_0 consistent with kink stability.

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Fig. 1. POPCON FOR ITER ASSUMING $Z_{\text{eff}} = 1.5$, H-MODE.
 MURAKAME DENSITY LIMIT, $n_{Mu} = 0.91 \times 10^{20} \text{m}^{-3}$,
 GREENWALD DENSITY LIMIT, $n_{Gr} = 1.67 \times 10^{20} \text{m}^{-3}$.

N-Weighted Volume Averaged Temp., T_e (keV)

Fig. 2 POPCON For ITER Assuming $Z_{eff} = 1.5$, L-Mode

August 20, 1988

ITER-88-053

MEMORANDUM

To: Distribution

From: R.B. Campbell, R.S. Devoto, and M.E. Fenstermacher

Subject: ITER Steady State Operating Space for Three Energy
Confinement Scaling Laws.

SUMMARY AND CONCLUSIONS

The purpose of this memo is to illustrate the effect that different confinement scaling laws have on the steady-state performance of ITER. We assume that the current is driven by a combination of neutral beams, lower-hybrid waves, and neo-classical "bootstrap" effects. The geometry of the ITER we examine is the 5.8 meter major radius case "BDE" currently being considered by the ITER group in Garching ($R_0 = 5.8$ m, $a = 2$ m, $\kappa_{95} = 2$, $B_0 = 5.1$ T). The scaling laws we consider are: 1) Goldston L-mode with an enhancement factor $H = 1.5$, 2) JAERI, with an H factor of 1.5; and 3) Rebut-Lallia L-mode confinement ($H = 1.0$). An attached Appendix presents the functional forms of the confinement laws we use. Plots of important parameters in plasma current, I_p , versus Troyon coefficient, C_T , space (SOPCONS)¹ are the primary means for the comparison. From these contour plots, reasonable operating points can be selected. A table of operating point parameters obtained from these SOPCONS as well as the SOPCONS themselves are included in this memo.

The primary conclusion of this memo is that attractive current-driven machines can be defined which obey the latest ITER rules, if either Goldston L-mode with H of 1.5 or Rebut-Lallia L-mode confinement applies to ITER. If L-mode JAERI scaling with $H = 1.5$ prevails, the performance of a steady state ITER must tolerate a large current drive power (≥ 150 MW)

¹R.S. Devoto, M.E. Fenstermacher to be published.

for machines providing a reasonably high wall loading $\approx 1 \text{ MW/m}^2$. If full H-mode ($H=2.0$) is postulated for JAERI scaling, the performance improves dramatically, to the point where it is comparable to the L-mode Goldston with $H = 1.5$ and L-mode Rebut-Lallia cases. Table 1 shows, for the three scaling laws, a comparison of some typical operating points.

Using a model by Barr² for the divertor heat flux, the limit of $\hat{H}_{h,div} \leq 5 \text{ MW/m}^2$ is satisfied for all cases, where $\hat{H}_{h,div}$ is the peak divertor heat flux. This model is presently being benchmarked to D-III data and appropriate runs of the Braams code. It should be noted that this model disagrees with Harrison's divertor model which is currently being used by the ITER team in Garching. Barr's model suggests that the heat load for a fixed amount of current drive power and alpha particle power is less than what Harrison would predict.

SPECIFICS

Steady-state Operation CONTOurs (SOPCONS) facilitate the comparison of current-driven machines operating with different plasma current, I_p , and plasma toroidal beta, β_t . The plasma beta in these plots is varied by changing both the plasma current and the Troyon coefficient, C_T , through the relationship $\beta_t = C_T \frac{I_p}{aB_0}$. By using these plots, we can define attractive operating regimes, as well as examine the gradient of key operating parameters in the $(I_p - C_T)$ space. When guidelines governing the minimum or maximum values of operating parameters have been chosen, windows of machine operation can be drawn.

The figures of merit we have chosen for display on the SOPCONS are the current drive power, P_{cd} (MW), the average neutron wall loading, $\langle \Gamma \rangle$ (MW/m^2), the safety factor at the 95% flux surface, q_{95} , plasma gain, Q , and the neutral beam energy, E_{inj} (MeV). The beam energy is set from penetration considerations, and scales with the line averaged density, $\bar{n}_{e,2\ell}$, Z_{eff} , path length, L , for an assumed number $\tau_b \approx 2$, of e-foldings of the beam intensity to the center of the plasma. The scaling is:

$$E_{inj} \approx 0.82 \times \left(\frac{\bar{n}_{e,2\ell} L Z_{eff}}{2\tau_b} \right)^{\frac{1}{3}}, \quad (1)$$

²W.L. Barr Paper 37-02, 12th Symposium on Fusion Engineering Monterey, Oct. 1987, and TIBER/ETR Final Design Report UCID-21150, Volume 1, p2-115 September 1987.

which comes from the scaling of the effective cross section for ionization with beam energy.

In the SOPCON plots, we have avoided drawing windows of operation, since the guidelines are still largely undefined. For example, the terms of reference call for $\langle \Gamma \rangle$ to be 1 MW/m² or greater, although it has been proposed that for continuous operation a minimum $\langle \Gamma \rangle$ closer to 0.7 MW/m² might be acceptable. The plots will show that this difference alone makes a substantial difference in the size of possible operating windows, allowing operation at lower beta and current drive power.

A total of six SOPCONs are shown in Figures 1-6, two plots for each scaling law. The value for Z_{eff} is held fixed at 1.78, and the fraction of the total current attributable to the bootstrap effect is fixed at 30%. There is a single impurity ion considered, with a $Z = 10.67$ to model the mix of various impurities chosen in the ITER study. The recommendation is that carbon and oxygen each comprise 47% of the total impurity density, iron makes up 4.7% of this density, and tungsten makes up slightly under 1%. The parameter which is changed to generate the two plots for each law is the density profile exponent, α_n . Both the density and temperatures are assumed to have roughly the following dependence on the radial coordinate:

$$X\left(\frac{r}{a}\right) = X_0 \left[1 - \left(\frac{r}{a}\right)^2 \right]^\alpha, \quad (2)$$

The density profile exponent, α_n , is either set to 1.0 or 0.5, to represent operation with peaked or flat density profiles, respectively. The flat density profile ($\alpha_n = 0.5$) is currently the guideline for the ITER study. To ensure the MHD stability will be roughly the same in the two cases, the temperature exponent is adjusted so that the radial profile of the thermal pressure is constant. The primary effect of flattening the density profile is to decrease the fusion power, thereby changing the plasma power balance point to one at which the average densities are higher and temperatures lower. The higher densities and lower temperatures result in less efficient current drive, thereby increasing the power required to drive a given current.

Figures 1 and 2 show the SOPCON space for Goldston L-mode scaling with an enhancement factor of 1.5 for the two density exponents. For both density profiles, attractive cases can be obtained with energies at or below $E_{inj} = 1.6$ MeV, average wall loadings above 0.8 MW/m², and current drive

powers at or below 100 MW. The plasma Q value is above 7.5 for this range of parameter space. The density for power balance in the hatched region exceeds the Greenwald limit.

Rebut-Lallia L-mode scaling produces machines which are even more attractive. Note that the scaling prescription consists of a *sum* of times, $\tau_{RL} = \tau_R + \tau_L$. For ITER parameters, τ_L is on the order of 1 – 2 seconds, whereas τ_R is only 10's of *milliseconds*. Figures 3 and 4 show the SOPCON analysis for the Rebut-Lallia scaling. Even for the flatter density profiles, the current drive power ranges from about 80 to 100 MW over the entire $(I_p - C_T)$ space. It is in fact quite easy to achieve average wall loadings in excess of 1 MW/m², and keep the power and beam energy requirements reasonable, without exceeding the peak divertor heat flux limit.

Figures 5 and 6 show what happens with the ITER "BDE" when the L-mode JAERI scaling with $H=1.5$ is utilized. The most pronounced feature is the high current drive power required ($P_{CD} > 150$ MW), regardless of the density profile assumed. In principle, the Troyon coefficient could be reduced below the minimum $C_T = 0.02$ shown on the plot to lower the current drive power, but the wall loading would be substantially reduced, as would the plasma Q value. Beam energies required for this scaling are somewhat above the 1.5 MeV range for this size device.

Table 1 compares some of the parameters for the three scaling laws with $\alpha_n = 0.5$. To define a representative operating point, we have taken the constraint that $\langle \Gamma \rangle \geq 1$ MW/m². For the JAERI case, we require that Q be as large as possible for $C_T \leq 0.03$. For the other two laws, we require that the beam energy stay below 1.6 MeV and the current drive power not exceed 100 MW. Each of these criteria define a region in $I - C_T$ space, from which we pick a central point as the typical case. The bootstrap current, I_{bs} , comprises 30% of the total current. The pressure profile is held fixed when α_n is changed. In this table, we display the Troyon coefficient, C_T , the plasma current, I_p , the volume-averaged ion and electron temperatures, T_i and T_e , the volume-averaged electron density, n_e , the Greenwald density limit, n_{Gr} , the average neutron wall loading, $\langle \Gamma \rangle$, the neutral beam injection energy, E_{inj} , the plasma gain factor, Q , efficiencies for neutral beam and lower hybrid current drive, γ_{nb} and γ_{lh} , ($\gamma_z = \langle n_z \rangle I_z R_0 / P_{inj}$, where $z = nb, lh$), the respective powers required for beams and lower hybrid, P_{nb} and P_{lh} , the fusion power, P_{fus} , and various β values. The thermal beta values are denoted by a subscript "th" for both

Parameter	Goldston	Rebut-Lallia	JAERI	JAERI H=2
C_T	0.0266	0.0295	0.021	0.03
I_p (MA)	21.5	20.75	21.5	20.0
Q	10.3	11.7	6.3	11.8
$\langle T_n \rangle$ MW/ πa^2	1.12	1.01	1.02	1.24
$\dot{H}_{h,div}$ (MW/m ²)	3.8	3.24	4.2	3.9
P_{CD} MW	103.5	82.3	155.5	100.0
P_{fus} MW	1067.0	962.0	974.0	1177.0
E_{inj} MeV	1.63	1.38	2.0	1.73
$T_e(T_i)$ keV	19.3 (18.0)	21.8 (22.1)	14.5 (12.2)	19.3 (18.1)
n_e 10 ²⁰ m ⁻³	0.63	0.56	0.75	0.66
n_{Gr} 10 ²⁰ m ⁻³	1.32	1.27	1.32	1.23
$\gamma_{nb}(\gamma_{th})$ (10 ²⁰ A/W-m ²)	0.55 (0.51)	0.58 (0.55)	0.43 (0.42)	0.55 (0.52)
$P_{nb}(P_{th})$ MW	76.2 (27.3)	63.3 (19.0)	105.5 (50.0)	73.5 (26.5)
$\beta_{th,e}(\beta_{th,i})$ %	2.3 (1.8)	2.3 (2.0)	2.0 (1.5)	2.4 (1.3)
$\beta_\alpha(\beta_{hot,i})$ (%)	1.2 (0.3)	1.5 (0.3)	0.6 (0.3)	1.3 (0.3)

Table 1: Scaling Law comparison, $Z_{eff} = 1.78$, $f_h = 0.3$, $\alpha_n = 0.5$. Goldston and JAERI are L-mode with $H = 1.5$, and Rebut-Lallia is L-mode ($H = 1.0$). Volume averaged densities and temperatures are reported.

electrons and all ions. The hot alpha beta is β_α , and the fast ion beta due to the neutral beams is $\beta_{hot,i}$. The peak divertor heat flux, $\hat{H}_{h,div}$, is also shown on this table as calculated from Barr's model. The fraction of the power radiated in the core plasma and scrapeoff layer is assumed to be 30%

The message from Table 1 is that steady-state ITER operating points can be found for either Goldston L-mode ($H=1.5$) or Rebut-Lallia L-mode scaling, which obey the current ITER rules. Using Barr's model for the peak divertor heat flux, the limit of 5 MW/m^2 is not violated for all three laws. A memo is currently under preparation by Barr showing the benchmark of this model to D-III-D data and Braams code calculations. The basic model is described in the 1987 TIBER report³. The scaling of the peak divertor heat flux is given by:

$$\hat{H}_{h,div} = \frac{(1 - f_{rad})^{\frac{1}{2}} 0.8 \left(0.2 + \frac{1}{Q}\right) P_{fus}}{A_{div}} \quad (3)$$

The factor of $\frac{1}{2}$ is for upper/lower divertor symmetry, and the factor of 0.8 is to account for the fraction which is intercepted by the inner divertor plate. The parameter f_{rad} is the fraction of the total power loss which is radiated from the plasma. It is the sum of a fraction (typically 10%) which is radiated from the core plasma, and a fraction which is radiated from the scrapeoff layer (assumed to be 20%). The scaling for the divertor area is:

$$A_{div} = 1.9 \times (R_0 + a) a \sqrt{\frac{\kappa B_{T,0} R_0}{I_p (R_0 - \delta a)}} \quad (4)$$

The variables κ and δ are the edge elongation and triangularity respectively.

The use of the L-mode JAERI scaling with $H=1.5$ results in higher current drive powers, and higher beam injection energies. The primary reason for the degradation in performance in this case is the relatively low temperatures and high densities required to obtain a plasma power balance. Under these conditions, the current drive efficiencies are somewhat lower than for the solutions with the other two scaling laws. If one postulates

³W.L. Barr Paper 37-02, 12th Symposium on Fusion Engineering Monterey, Oct. 1987, and TIBER/ETR Final Design Report UCID-21150, Volume 1, p2-115 September 1987.

that full H-mode can be obtained, and uses an H-factor of 2, the values in the last column of Table 1 are obtained. The improvement in performance is sufficient to make the operating point comparable with those from the other scaling laws.

APPENDIX- Functional forms for Energy Confinement Scaling Laws

Most of the parameters in the scaling laws have been defined in the body of this memo, unless otherwise noted.

The energy confinement time for the L-mode Goldston Law can be expressed:

$$\tau_G = 0.037 I_p P_{heat}^{-0.5} R_0^{1.76} a^{-0.37} \kappa_{95}^{0.6} A_{DT}^{0.5} \quad (5)$$

Where P_{heat} is the net heating power to the plasma (alpha, auxiliary, minus radiation), and A_{DT} is the effective ion mass in AMU, $\frac{A_\alpha + A_x}{5}$.

The Rebut-Lallia scaling law takes the form⁴:

$$\tau_{RL} = 0.052 \bar{n}_{e,19}^{0.76} Z_{eff}^{0.25} B_0^{0.5} I_p^{0.5} P_{heat}^{-1.0} + 0.024 I_p^{1.0} L_R^{1.5} Z_{eff}^{-0.5} \quad (6)$$

The variable $\bar{n}_{e,19}$ is the line-averaged electron density in units of 10^{19} m^{-3} . The parameter L_R is given by:

$$L_R = (R_0 a^2 \kappa)^{\frac{1}{3}} \quad (7)$$

The κ without the subscript is measured to the separatrix.

Finally, the JAERI law (also known as the Odajima-Shimomura law) obtained from JT-60 data is given by:

$$\tau_J = 0.085 \kappa a^2 A_{DT}^{0.5} + 0.069 \bar{n}_{e,20}^{0.8} I_p B_0^{0.2} \kappa^{0.2} a^{0.4} R_0^{1.6} P_{heat}^{-1.0} A_{DT}^{0.5} g_J \quad (8)$$

The function g_J is given by:

$$g_J = Z_{eff}^{0.4} \left(\frac{15 - Z_{eff}}{20} \right)^{0.6} \left[\frac{3q_{*,0}(q_{*,0} + 5)}{(q_{*,0} + 2)(q_{*,0} + 7)} \right]^{0.6} \quad (9)$$

The variable $q_{*,0}$ is a safety factor defined in this context by the relation:

$$q_{*,0} = \frac{5a^2 B_0}{2I_p R_0} [1 + \kappa^2 (1 + 2\delta_*^2 - 1.2\delta_*^3)] \quad (10)$$

The parameter δ_* is the triangularity at the separatrix.

⁴A. Gibson, *et. al.* ; 15th European Conference on Plasma Physics and Controlled Fusion Cavity, May 1988 quoting Rebut-Lallia from Varenna 1987

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PLASMA CURRENT, I_p (MA)

Figure 2-Goldston enhanced L-mode SOPCON, $\alpha_n = \frac{1}{2}$, $\alpha_T = 1$, $H = 1.5$.

PNRARR: MISTAKE, N. 5.00m, 2.00m, K. 2.0, d. 4.00, 5.10T, H. 1.00

aln = 1.0, alt = .5, Eb = 0, MeV, Fa = 1.00, Zeff = 1.75

Figure 3-Rebut-Lalia L-mode SOPCON, $\alpha_n = 1$, $\alpha_r = \frac{1}{2}$, $H = 1.0$.

Figure 4-Rebut-Lallia L-mode SOPCON, $\alpha_n = \frac{1}{2}$, $\alpha_T = 1$ $H = 1.0$.

Figure 6-JAERI enhanced L-mode SOPCON, $\alpha_n = \frac{1}{2}$, $\alpha_T = 1$, $H = 1.5$.

August 24, 1988

To: Distribution
From: R.B. Campbell
Subject: Neutron Wall Loading Limits Based on Heat Load Considerations.

SUMMARY AND CONCLUSIONS

Different models can and do predict significantly different peak heat fluxes on the divertor plates. It is suggested that the appropriate decisionmakers take into consideration this apparent disagreement between the Harrison and Barr models before any programmatic and/or configurational decisions are finalized. It is also suggested that there be an improvement in communication between the different groups promoting different divertor heat flux models. This would improve all the models as well as point out the strengths and weaknesses of each treatment. There is also a critical need to benchmark these models to *experimental* data from appropriate machines.

In a recent set of workshop notes¹, the work due to Vieider and Putvin-skij concluded that for a current driven steady-state machine to operate below $\bar{H}_{div} = 5 \text{ MW/m}^2$ for the peak heat flux on the divertor plates, the neutron wall loading could not exceed $\langle \Gamma_n \rangle = 0.3 \text{ MW/m}^2$. Their analysis relies, to the best of our knowledge, on the divertor heat flux model of Harrison². Furthermore, we believe a limit for first wall heat flux at a neutron wall load of $\langle \Gamma_n \rangle = 1 \text{ MW/m}^2$ was taken to be $\bar{H}_{wall} = 0.2 \text{ MW/m}^2$ at 10^6 cycles in their analysis. This latter limit puts a restriction on the amount of power which can be radiated in the scrapeoff layer.

¹"Preliminary Study on Phased Operation with Flexibility" presented by Y. Shimomura at, ISTAC Meeting, July 13, 1988

²M.F.A. Harrison, "Control of Plasma Impurities and the Exhaust of Power and Helium" Notes from ITER Workshop, April 1988. Also Appendix 5 in Preliminary NET report November 1988.

A model for the peak divertor heat flux, $\bar{H}_{a,div}$, been recently advanced by Barr³ which includes the detailed variation of the poloidal and toroidal magnetic fields and their impact on the distribution of the heat deposition on the plates. Barr's model predicts a somewhat lower peak heat flux for a given amount of exhaust power than does Harrison's model. We use Barr's model for predicting the peak divertor heat flux in this memo. Barr's model is being benchmarked against D-III data and compared with the Harrison model, and results will be reported soon.

Using the curves presented at the workshop giving first wall surface heat flux limits versus number of cycles, the allowed wall heat flux at $\langle \Gamma_n \rangle = 1$ MW/m² could be increased to about 0.4 MW/m² if the number of burn pulse cycles is reduced a factor of 10. We expect this kind of reduction in cycles going from a pulsed machine to a (quasi-) steady-state device with current driven non-inductively.

Using Barr's model, We also find that the divertor heat flux problem is certainly formidable. However, We also find that steady-state operating scenarios at $\langle \Gamma_n \rangle \approx 1$ MW/m² can be defined which do not exceed either the peak divertor heat flux of 5 MW/m², or the $\bar{H}_{wall} = 0.4$ MW/m² average wall heat flux limits. Figure 2 shows this clearly.

We also demonstrate (Figure 4) that with Goldston enhanced L-mode confinement, and Barr's model for the peak divertor heat flux, a neutron wall loading of up to 1.6 MW/m² can be achieved in a steady state current driven machine.

DETAILS

We first examine the results of the Garching Team, based on Harrison's model for the peak divertor heat flux. Figure 1 shows the minimum plasma power gain, $Q (= P_{fus}/P_{inj})$, allowed to stay below a specified heat load limit as a function of the fraction, α_{div} , of the exhaust power which impinges on the divertor plates. The remaining power assumed radiated in the scrapeoff layer and deposited on the first wall, with a heat flux $\bar{H}_{wall}$. Several wall loadings are shown on this plot to illustrate the fact that as the wall loading goes up, the window for allowed operation shrinks. The curve to the left of the minimum represents the radiation heat flux limit on the first wall of

³W.L. Barr, Paper 37-02, 12th Symposium on Fusion Engineering Monterey, Oct. 1987

approximately 0.2 MW/m^2 , the right part of the curve reflects the limit of 5 MW/m^2 on the peak divertor heat flux, $\dot{H}_{div}$.

Notice that for the nominal $\langle \Gamma_n \rangle = 1 \text{ MW/m}^2$, at $Q = 10$ the range of α_{div} is reasonably wide, whereas if ITER falls short of the $Q = 10$ performance, say $Q = 5$, the window of operation disappears. For a different set of assumptions which take into account important effects we can demonstrate that the window of operation remains open even at the lower Q values.

The methodology we use can be summarized below:

- We use the Barr model for $\dot{H}_{div}$. This model includes effects of the magnetic mapping from the torus midplane to the divertor plates. The model also includes the radial variation of the pathlength along a field line a particle must travel before impinging on the divertor plate. Both of these details have an important role in determining the effective divertor area and the spreading out of the heat load. Use of Barr's model shifts the curves of minimum Q downward, opening the window of operation.
- We assume a specified confinement law and L-mode enhancement factor, H . We pick the Goldston law with $H = 1.5$. The Garching team analysis assumes the confinement time will be whatever is necessary to produce the losses to give the minimum Q value. We also vary the value of the confinement time, but we do it by varying the plasma current, I_p , and the Troyon coefficient, C_T . Of course, lower I_p results in a lower confinement time and Q . For this task, the SOPCON analysis⁴ can be quite useful, because it illustrates how the heat fluxes, neutron wall loadings, Q values vary as I_p and C_T are changed. The distinction between the two ways to treat confinement may seem inconsequential at first, however since the effective divertor area scales from the Barr model as $A_{div} \propto 1/\sqrt{I_p}$ for a constant geometry, the curves of Q_{min} at constant $\langle \Gamma_n \rangle$ will shift to lower Q , thereby further widening the operating window.
- We assume that at $\langle \Gamma_n \rangle = 1 \text{ MW/m}^2$ the first wall heat flux can be increased to $\dot{H}_{wall} = 0.4 \text{ MW/m}^2$. This is justified for a steady-state current driven machine, since the number of cycles during machine

⁴R.S. Devoto, M.E. Fenstermacher, to be published

nuclear testing operation will be reduced compared to a pulsed machine by a factor of from 5 to 20.

Figure 2 shows our results for the minimum Q required as a function of the fraction of the heat to the divertor plates. Only the $\langle \Gamma_n \rangle = 1 \text{ MW/m}^2$ case is considered here, since it was judged to be the most important one. For reference, the Garching result for the same neutron wall load has been re-plotted as well.

The main point to be made here is that a combination of Barr's divertor model and assumptions appropriate for a steady-state machine governed by a given confinement scaling law results in a wide window of operation, even down to Q values (≈ 2.5) which would be unacceptable for reasons unrelated to the heat load issue. Figure 3 shows the plasma current and Troyon coefficient required to generate the Figure 2 $Q_{\min}$ curve. Notice from this curve that the plasma currents to achieve $Q_{\min}$ are substantially below ranges of 20-30 MA currently being considered. This is merely a statement that the Q values we expect from this size ITER with Goldston $H=1.5$ scaling will be larger, typically around 10.

Figure 4 demonstrates what happens to the *allowed* neutron wall loading when the fraction of the exhaust power which impinges on the divertor plates is varied. In this scan, the plasma current is fixed at 20 MA. The Q value for this machine with Goldston scaling with $H = 1.5$ is about 10 for all cases. As is to be expected, the curve is governed by the radiation heat flux limit at low values of α_{div} , and by the peak heat flux condition at high α_{div} . At intermediate α_{div} , the requirement that C_T not exceed 0.03 limits the wall loading to about 1.6 MW/m^2 in this regime.

This figure shows that with Goldston enhanced L-mode confinement, and Barr's model for the peak divertor heat flux, a neutron wall loading of up to 1.6 MW/m^2 can be achieved in a steady state current driven machine.

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Figure 1- Q_{min} versus α_{div} for several neutron wall loadings. The Harrison Model for the peak divertor heat flux is used in this analysis.

by S.V. Putviniskij

Figure 2- Q_{min} versus α_{div} for an average neutron wall loading of $\hat{\Gamma}_n = 1 \text{ MW/m}^2$. Curves are shown for both the Harrison Model and the Barr Model for the peak divertor heat flux.

Figure 3-Plasma current, I_p , and Troyon coefficient C_T , versus α_{div} for the Q_{min} curve of Figure 2 for the Barr model and Goldston H=1.5 scaling.

Figure 4-Allowed average neutron wall loading, $\bar{\Gamma}_n$, versus α_{div} for a fixed plasma current $I_p \approx 20$ MA, Goldston H=1.5 scaling. Barr model for $\bar{H}_{div}$ is used.

August 26, 1988

BENCHMARK OF TETRA AGAINST MUMAK-Fenstermacher/Campbell

Notes:

- Full hot ion pressure used in beta limit.
- Benchmarking TETRA case "33.8c".
- Uckan's model for $\beta_{\alpha,hot}$ had to be used in MUMAK to get agreement.
- Formula for bootstrap current needs to be resolved. Nevins vs. Devoto formula, use of thermal vs. total β_p . MUMAK was run with TETRA's computed bootstrap fraction of 0.41.
- MUMAK calculation of the poloidal field should be changed. Approximation of perimeter for β_p in MUMAK results in lower β_p .
- Synchrotron radiation powers do not agree. This is because the wall reflectivity was taken to be 0.9 in MUMAK and 0.85 in TETRA. A rerun of MUMAK with 0.85 resulted in slightly better agreement than below.
- Lower hybrid current drive efficiency does not agree very well.

General Input Parameters:

$$I_p = 18.6 \text{ MA}$$

$$Z_{eff} = 1.78$$

$$H = 2.07, \text{ Goldston}$$

$$B_0 = 5.1 \text{ T}$$

$$C_a = 0.0538$$

$$C_T = 0.03$$

$$\kappa_1 = 2.2$$

$$\delta_1 = 0.6$$

$$\kappa_{95} = 1.91$$

$$\delta_{95} = 0.4$$

$$\alpha_n = 0.5$$

$$\alpha_T = 1.0$$

Parameter	MUMAK	TETRA
$T_e(T_i)$ keV	19.88 (19.375)	19.91 (19.381)
n_e 10^{20} m $^{-3}$	0.576	0.5699
β_e (β_i) %	2.115 (1.815)	2.115 (1.815) approx.
$\beta_{a,hot}$ (β_{beam}) %	1.179 (0.361)	1.179 (0.357)
$\langle \Gamma \rangle$ MW/m 2	0.932	0.9
Q	15.7	15.8
P_{sync} MW	19.42	24.01
P_{brem} MW	17.00	16.71
E_{beam} MeV	1.31	1.32
η_{nb} A/W	0.1847	0.185
η_{lh} A/W	0.1517	0.1815
I_{nb} MA	8.59	8.24
I_{lh} MA	2.36	2.69
τ_E sec	2.667	2.828
$f_{a,c}$	0.608	0.618
$f_{nb,i}$	0.4692	0.46
P_{CD} MW	61.87	59.39
P_{lh} MW	15.56	14.81
P_{nb} MW	46.55	44.57
P_{fus} MW	971.	938.

A STEADY STATE TOKAMAK REACTOR BASED ON ITER*

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ABSTRACT

The physics and engineering guidelines for the ITER device are shown to lead to viable and attractive operating points for a steady state tokamak power reactor. Non-inductive current drive is provided in steady state by high energy neutral beam injection in the plasma core, lower hybrid slow waves in the outer regions of the plasma and bootstrap current. Plasma gain Q (=fusion power/input power) in excess of 20 and average neutron wall loading, $\langle \Phi \rangle \approx 2.0$ MW/m² are predicted in a device with major radius, $R_0 = 7.5$ m and minor radius, $a = 2.8$ m.

INTRODUCTION

The terms of reference [1] for the International Thermonuclear Experimental Reactor (ITER) state that ITER should demonstrate "technologies essential to a reactor in an integrated system." In addition, ITER "will provide the data base in physics and technology necessary for the design and construction of a demonstration fusion power plant." Part of the process of connecting ITER with a demonstration power plant is to perform parametric variations of steady state reactor configurations using the same current drive scenarios envisaged for ITER. At present, high energy negative-ion neutral beam injection (NBI) for current drive in the plasma core supplemented with lower hybrid (LH) slow wave current drive on the outer flux surfaces and bootstrap current, is the favored current drive scenario for the ITER design in steady state. The goal of this paper is to show that this current drive concept leads to plasma conditions appropriate for a viable power producing reactor.

The neutral beams must drive current in the plasma core both to furnish the seed current

necessary to produce bootstrap current and because the LH waves can not penetrate to the axis in a reactor grade plasma. To assure penetration, the neutral beam energy, E_{nb} , is chosen to yield about a five e-fold reduction in beam current along the full beam path through the plasma. Injection tangential to the magnetic axis is used to maximize the current drive efficiency while maintaining sufficient current drive near the axis. Lower hybrid waves propagate only part of the way into the plasma from the outer edge because of a combination of an accessibility condition (primarily a function of plasma density) and a condition for avoiding strong quasi-linear electron Landau damping of the wave (a function of the plasma temperature). For reactor grade plasmas with peak electron density, $n_0 = 10^{20}$ m⁻³, and temperature, $T_e = 25$ keV, the LH waves penetrate only to a normalized minor radius $r/a \approx 0.7$.

The remainder of this paper is organized as follows. In Section II we present the physics and engineering relations which limit the available parameter space for acceptable reactor operating points. We describe a convenient method for examining possible operating space subject to these constraints in Section III. Results are presented in Section IV including trade-offs in device performance for various operating points and the effect of various assumptions on the available operating space. Section V presents our conclusions.

CONSTRAINING RELATIONS

A fundamental limitation on plasma parameters is the critical toroidal beta beyond which the plasma is expected to become unstable to ideal ballooning or kink modes. This limit is conveniently expressed in terms of the Troyon formula [2]

$$\beta \leq \beta_T = C_T I_p / a B_0 \quad (1)$$

where I_p is the plasma current, a is the minor radius, B_0 is the toroidal field at the geometric major radius, R_0 , and C_T is a coefficient, typically 3.0, which sets the upper limit on β . MKS units are used throughout this paper except for current in kA and temperature in keV. In addition, theoretical studies of ideal kink stability show that the plasma is likely to be unstable to $n = 1$ kinks if the edge safety factor q_e drops below about 3.0. It should be noted that some experiments have been able to achieve stable operation at q_e approaching 2.5. The safety factor is given by

$$q_e = B_0 R_0 \oint (ds / R^2 B_p) \quad (2)$$

with B_p the local poloidal field, and R the major radial coordinate. The limit on q_e is typically applied where the poloidal flux is 95% of the total flux enclosed by the separatrix. The limit becomes $q_{95} > 3.0$. Since it is necessary to compute the 2-D MHD equilibrium to find the value of q_{95} , it is desirable to have a simple formula for safety factor. We use the form

$$q_e = \frac{5 a^2 B_0}{R_0 I_p} \frac{1 + \kappa_{95}^2 (1 + 2\delta_{95}^2 + 1.2\delta_{95}^3)}{2} \quad (3)$$

where κ_{95} is the elongation and δ_{95} is the triangularity of the 95% flux surface. Since q_e is typically less than q_{95} , we should take our limit lower than 3.0, but we do not yet have sufficient information to choose a precise number. There are additional conditions on κ_{95} and δ_{95} , typically $\kappa_{95} < 2.5$ and larger δ_{95} required at larger κ_{95} ($\delta_{95} = 0.5$ at $\kappa_{95} = 2.5$).

For the density limit we use the Greenwald limit [3], given by

$$\bar{n}_e \leq \bar{n}_{Gr} = f_{Gr} \kappa_{95} I_p / A_p \quad (4)$$

where $\bar{n}_e$ is the line-averaged density, A_p is the plasma cross sectional area and the coefficient f_{Gr} is taken as 1.0. We note that the density limit is taken to be a "soft" limit in the ITER design process. This is interpreted to mean that operation near this density does not, by itself, imply that the frequency of disruptions increases. In practice, this limit should seldom be approached in steady state operation.

A variety of scaling laws are possible for energy confinement time. We use one of the models currently used for the ITER studies, namely the relation proposed by Goldston [4],

$$\tau_E = 0.037 I_p^{-0.75} \kappa_{95}^{-0.5} P_n^{-0.5} a^{-0.33} (A_1 / 1.5)^{0.5} \quad (5)$$

where A_1 is the average plasma mass, and the total heating power P_h is the sum of the current-drive power and the alpha power less the radiated power.

It is widely believed that the energy-confinement time cannot exceed the Ohmic, or neo-Alcator time given by [5]

$$\tau_{RA} = 0.0625 \bar{n}_e a R_0^2 q_e \quad (6)$$

In practice, τ_{RA} is always greater than τ_E under predicted steady state conditions for reactor sized devices. To assure that the confinement time used in our study does not exceed τ_{RA} , we use the formula, $\tau_E = [\tau_{RA}^{10} + (H \tau_E^0)^{10}]^{-0.1}$ where H is described below.

The Goldston scaling was developed for L-mode discharges in limiter plasmas. It was recently noticed that L-mode discharges in divertor plasmas in JET [6] displayed somewhat better confinement time, perhaps as much as 50% higher than the value predicted by τ_E . For these studies we therefore compute τ_E using the prediction from Eq. (5) multiplied by a constant $H = 1.5$.

At present, the likely candidates for non-inductive current drive are lower hybrid (LH) slow waves in the outer region of the plasma, high-energy neutral beams (NB) in the core, and neoclassical (bootstrap) current. The LH waves are limited to the outer region because the choice of $N_{||}$ dictated by the accessibility condition means that the waves are strongly Landau damped on electrons in the outer plasma region where the temperature reaches 10-15 keV [7]. The accessibility condition is given by

$$N_{||} \geq N_{||ac} = \frac{\omega_{pe}^2}{\omega_{ce}^2} + (1 + \frac{\omega_{pe}^2}{\omega_{ce}^2} - \frac{\omega_{p1}^2}{\omega_o^2})^{1/2} \quad (8)$$

where ω_{pj} and ω_{cj} are the local plasma and cyclotron frequencies of species j and ω_o is the frequency of the launched waves. The condition for strong electron Landau damping can be written approximately as [7]

$$N_{||} \leq N_{||ELD} = 7 / T_e(r) \quad (9)$$

In our calculations of the power required for current drive, we choose $N_{||}$ at the radius of maximum penetration into the plasma, r_{max} , to satisfy both of these conditions. The required power is reported in the form of a current-drive figure of merit

$$\gamma_{LD} = n_{e20}(r_{max})^{1/2} I_p P_o / P_{LD} \quad (10)$$

where $n_{e20}(r_{max})$ is the electron density at r_{max} in units of 10^{20} m^{-3} and I_p and P are in kA and MW respectively.

lower hybrid driven current and absorbed power, respectively. To compute γ_{lh} we use the expression due to Fisch [8], multiplied by an adjustment factor to bring it into agreement with detailed calculations of ray tracing and power deposition [9]. This correction factor is typically 0.6.

For the NB figure of merit we employ an expression which can be obtained from an analytic expression for the fast ion distribution function [10]. The expression reads

$$\gamma_{nb} = \frac{\langle n_{e20} \rangle I_{nb} R_0}{P_{nb}} = \frac{C_{nb} \langle T_e \rangle J(\langle x \rangle, \langle y \rangle) \langle F_e \rangle}{\ln A_e} \quad (11)$$

where $\langle \rangle$ denotes a volume average and $J(x, y)$ is a function which takes into account the appropriate moment of the fast ion distribution function. It is approximated by [11]

$$J(x, y) = x^2 / \{4 + 3y + x^2(x + 1.39 + 0.69y^{0.7})\} \quad (12)$$

$x^2 = E_b/E_e$, $y = (\sum_i \ln A_i n_i Z_i^2 / A_i) / (\sum_i \ln A_i n_i Z_i^2 / A_i)$, A_b is the beam mass in amu, E_b is the NB energy, $\ln A_e$ and $\ln A_i$ are the Coulomb logarithms for electrons and ions respectively, and C_{nb} is a correction factor to fit the detailed calculations of beam deposition and fast ion current, performed with the ACCOME code [12]. $\langle F_e \rangle$ is the ratio of the net current to the total fast ion current, which accounts for the partial cancellation of the fast-ion current by the electrons. To calculate $\langle F_e \rangle$ we use a scaling formula [11],

$$\langle F_e \rangle = 1 - (Z_b/Z_{eff}) (1 - G(\langle \epsilon \rangle, Z_{eff})) \quad (13)$$

where

$$G(\langle \epsilon \rangle, Z_{eff}) = (1.55 + 0.85/Z_{eff}) \langle \epsilon \rangle^4 \cdot (0.2 + 1.55/Z_{eff}) \langle \epsilon \rangle \quad (14)$$

and $\langle \epsilon \rangle$ is an effective inverse aspect ratio. Detailed calculations [12] of NB current drive indicate that Eq. (13) should be evaluated with $\langle \epsilon \rangle = 0.46 a/R_0$.

The neutral beam energy is set by the requirement that the beams penetrate sufficiently close to the plasma axis to produce an MHD stable current profile there. From calculations of beam attenuation, including excitation-enhanced stopping cross sections [13], the following scaling formula has been developed for the beam energy required,

$$E_b (\text{MeV/amu}) = (1.37 \times 10^{-22} (Z_{eff} + 0.2)^{0.65})$$

$$[3 \bar{n}_b / (0.2 T_e + 1.0)]^{0.118} \bar{n}_b L / r_b \}^6 \quad (15)$$

where $\xi = (0.7 + 0.02 Z_{eff})^{-1}$, L is the beam path length from the plasma edge to the tangency point, and r_b is the number of e-fold reductions in beam current over the path L . For this study, we have used $r_b = 2.4$ in this formula based on our initial calculations of NB current drive.

The geometry of the tokamak reactor considered in this study is similar to the ITER design. The inner radial build includes a central OH coil stack, TF coil inner leg, and first wall + blanket + shield. The outer bore of the OH coil stack was constrained to be $R_{OH} \leq 2.0$ m. The elongation and triangularity on the 95% flux surface were set to $\kappa_{95} = 2.0$ and $\delta_{95} = 0.4$ respectively. A central toroidal field, $B_0 = 6.0$ T, was chosen based on a target constraint of about 14 T maximum field on the TF inner leg, although maximum field by itself is not regarded as a fundamental constraint. Fixed parameters for our reference device are given in Table 1.

Table 1. Reference case parameters.

$R_0 = 7.5$ m	$B_0 = 6.0$ T
$a = 2.8$ m	$\alpha_n = 1.0$
$\kappa_{95} (\kappa_p) = 2.0 (2.2)$	$\alpha_z = 0.5$
$\delta_{95} (\delta_p) = 0.4 (0.6)$	Deuterium beams

Another part of the radial build is the inboard first wall + blanket + shield and their associated manifolding and gaps. The thickness of these components is a function of the blanket type used. Based on work summarized in [14], this thickness varies between 1.07 m and 1.28 m over a range of blanket types. The thinnest case is for a water-cooled LiAlO_2/Be blanket while the thickest case is for a Li self-cooled blanket. For this work we will employ a He-cooled Li_2O blanket with a first wall + blanket + shield total inboard radial build thickness of 1.16 m.

The TF and PF magnet systems have been designed on the basis of several engineering constraints that ensure their reliability, protection, and structural integrity. These include, sufficient current and temperature margins of the conductor in the high field regions of the coils for good stability, an upper limit on the hot-spot temperature in a coil during quench and dump of 150 K (with 20 kV

Figure 2. SOPCON for the baseline case.

space corresponds to a unique value of beta through the Troyon formula. This uniquely determines the plasma density at power balance. These equations and the other physics expressions described above are incorporated into the MUMAK code [15].

Radial profiles of the density and temperature are taken to be of the form, $X(r) = X(0) [1 - (r/a)^2]^n$, where $x = n, T$. Power balance terms in MUMAK are integrated over these model profiles using a circular flux surface model to yield profile correction factors. The volume of the fully shaped plasma is used to compute the magnitude of the power balance terms.

RESULTS AND DISCUSSION

Fixed parameters for our reference device are given in Table 1. The major radius and aspect ratio are consistent with the ITER engineering design guidelines. Optimization of the device geometry for reactor performance or minimum cost has not been done. The SOPCON for this reference case is shown in Fig. 2. Goldston energy confinement scaling with an enhancement factor, $H = 1.5$ (limited by the Neo-Alcator confinement time), has been used. The shaded region shows where the density for power balance exceeds the Greenwald density limit. The density and temperature profile exponents are $\alpha_n = 1.0$ and $\alpha_T = 0.5$ respectively (the same pressure profile as in ITER), and the plasma effective charge is taken to be $Z_{eff} = 1.5$ with oxygen impurity. We have adopted the interim ITER guideline for bootstrap current fraction, $I_{bs}/I_p = 0.3$, though higher fractions may be possible in scenarios with high poloidal beta ($\beta_p \geq 1.0$).

Contours shown on the SOPCON include $Q = 20$ and 25, $P_{CD} = 110$ and 150 MW, $q_{95} = 4.0$ and 4.5 (scaled from the MHD calculation at $I_p = 25$ MA shown in Fig. 1), $E_b = 1.8$ and 2.4 MeV, and $\langle \Gamma \rangle = 1.5$ and 2.0 MW/m². Several possible operating points could be selected. For fixed neutron wall loading, $\langle \Gamma \rangle = 2.0$ MW/m², the operating point with the lowest current is located at $I_p = 23$ MA, and $C_T = 2.68$. The lower limit on the current in this case is due only to the soft density limit. The plasma Q is 21 at this point, with a current drive power requirement, $P_{CD} = 170$ MW, and a beam energy requirement, $E_b = 2.7$ MeV. Alternatively, a high current operating point with the same wall loading exists at $I_p = 36$ MA and $C_T = 3.08$. The plasma gain at this point is $Q = 28$ with current drive requirements, $P_{CD} = 120$ MW and $E_b = 1.4$ MeV. In this case the upper bound on the current, for fixed wall loading, is set by the beta limit. A more detailed list of parameters for an intermediate point are given in Table 3.

We note that the safety factor, at 95% flux, for the operating point in Table 2 is quite a bit higher than in the ITER design ($q_{95} = 5.3$ compared with $q_{95} = 3.2$ in ITER). Since ripple induced losses of fast alphas scale strongly with the safety factor, the use of ferromagnetic inserts might be necessary to reduce the ripple for low current operation. Our calculations have not included fast alpha losses, under the assumption that the required low level of ripple can be incorporated in the design.

The SOPCON shown in Fig. 3 was generated with the same assumptions as for Fig. 2 except that both the density and temperature (and therefore the pressure) profiles were assumed to be more peaked, $\alpha_n = 2.0$ and $\alpha_T = 1.0$. This shows the improvement in reactor performance which could be achieved if methods for central fueling and heating were fully developed. For the same intermediate current, $I_p = 25$ MA as in Table 2, and Troyon coefficient, $C_T = 2.68$ (which may be close to the limit for this pressure profile), the plasma Q value increases to $Q = 37$ with the corresponding decrease in current drive power, $P_{CD} = 90$ MW at an average wall loading, $\langle \Gamma \rangle = 2.0$ MW/m². The beam energy requirement is also somewhat lower, $E_b = 1.7$ MeV, from the scaling of Eq (15).

CONCLUSIONS

The steady state operating scenario for ITER leads to viable and attractive operation points for a tokamak power reactor. An operating point has been identified, below the beta and density limits, in a device with $R_0 = 7.5$ m and $a = 2.8$

terminal voltage during dump), and maximum combined stresses at any point in the case or windings which are within allowables for the materials involved. Table 2. summarizes features of the TF system for the present study. It is a direct, non-optimized extrapolation of present ITER design guidelines. The maximum von Mises stress estimated in the use region of the wedged TF cases is 670 MPa.

Table 2. TF System Parameters

Number of coils	16
Total current	225 MA
Winding pack geometry	
Cross-section (approx.)	(1.15 m x 0.61 m)
Mean radius of the inboard leg	2.9 m
Mean radius of the outboard leg	13.0 m
Mean height above the midplane	8.9 m
Total TF system stored energy	105.3 GJ
Maximum field at the windings	14.3 T
Winding pack current density	21.2 A/mm ²
Total TF system cold mass	10,400 t
Conductor descrip. (Nb ₃ Sn, cable-in-conduit)	
Operating current	32.9 kA
Critical current	54.8 kA
Current sharing temperature	6.2 K
Effective area of a turn	1552 mm ²
Area of steel in the conduit	666 mm ²
Area of insulation around conduit	154 mm ²
Area of conductor inside conduit	439 mm ²
Area of helium inside conduit	293 mm ²
Fraction of copper in conductor	0.67

The PF system is a closer extrapolation of the present ITER design since the critical central solenoid is essentially unchanged (1.4 m inner radius and 2.0 m outer radius). The present design provides a volt-sec capability similar to ITER but the total stored energy is higher, $\Psi_{TF} = 12$ GJ. The increase results from the increased radii of the divertor and equilibrium field solenoids. Design of the PF system is considerably eased by the dramatic reduction in fatigue cycles and ac losses in this steady state device, compared with ITER which must be designed to operate in pulsed ignited mode.

Combining these components of the radial build places the radius of the first wall on the

With this constraint, we chose a geometric major radius, $R_0 = 7.5$ m and a plasma minor radius, $a = 2.8$ m. The inner scrapeoff layer thickness is $\Delta_i = 0.2$ m. A typical set of equilibrium flux contours for this device operating with toroidal current, $I_p = 25$ MA, is shown in Fig 1.

Figure. 1. Equilibrium flux contours for the baseline case.

STEADY STATE OPERATING CONTOURS (SOPCON's)

We have found that plots of performance parameters (eg. Q = fusion power/current drive power, $\langle I \rangle$ = average neutron wall load, and current-drive power, $P_{cd} = P_{th} + P_{nb}$), in the I_p - C_r plane are convenient for comparing steady state operating points. We refer to these Steady state Operating Contour plots as SOPCON's. One such SOPCON, is shown in Fig. 2.

The power balances for electrons and ions are performed separately, including electron-ion drag power, for each point on the SOPCON. Alpha power and current drive power are balanced against transport losses and radiation power (Bremsstrahlung and synchrotron). Transport losses are calculated using empirical energy confinement scaling laws for τ_E . These equations determine the electron and ion temperatures. For a fixed configuration, each point in I_p - C_r

B Alfvén Wave Stability Problem

November 23, 1987

LLNL-ITER-88-012

MEMORANDUM

To: ITER Distribution

From: R.B. Campbell

Subject: ITER Current Drive Efficiency in the Presence of Alfvén Wave Turbulence

1 Summary

The baseline for ITER uses high energy neutral beam injection to drive a steady-state current. Beam penetration and standard current drive efficiency considerations lead us to consider energies in the range of 1-3 MeV. Apart from the ambitious extrapolation from present source and accelerator technologies, there is also concern that the high energy beams with speeds above the Alfvén speed ($V_A = \frac{B}{\sqrt{4\pi m_i n_i}}$) will excite normal modes of the plasma which could, through enhanced slowing down of the beam, degrade the current drive efficiency. In ITER, the Alfvén energy near the plasma center is about 420 keV.

The study plan is to use both analytic and numerical means to assess the importance of unstable compressional and shear Alfvén waves [1] on ITER current drive efficiency. We have already performed a cursory look at the density thresholds and growth rates for ITER using the results of Ref.[1], and have discovered that both the compressional and shear mode could be important.

For the compressional mode, the perpendicular width of the distribution function is an important parameter in determining the degree of instability of the mode. For the case of parallel injection of a beam above the energy where pitch angle scattering is important, the angular width of the source is important. Narrow widths are claimed to result in a more stable system,

according to Ref. [1]. For injection at an angle, as is the case for ITER, the perpendicular width of the distribution is more like $v_b \sin \theta_{inj}$, where v_b is the beam speed, and θ_{inj} is the injection angle. For ITER, using this width, the minimum beam density for the compressional mode to become unstable is about a factor of 25 lower than the beam density in the baseline. The growth rate of the mode is approximately 2000 sec^{-1} .

The shear mode also appears to be unstable for ITER. For a beam distribution plateau width which is consistent with a Fokker-Planck calculation [2], the density threshold for this mode is a factor of 2.6 lower than the beam density for ITER. The growth rate is quite large, on the order of one-tenth the real part of the frequency, $\gamma \approx 2 \times 10^5 \text{ sec}^{-1}$.

Recently, there has been experimental evidence [3] of MHD bursting activity in small tokamaks (PBX) at low density when beams are injected tangentially, resulting in radial losses of beam ions. Some of the mode's characteristics are similar to "fishbones" observed during perpendicular beam injection, which are thought to be a $m \approx 1$, $n = 1$ long wavelength form of the shear Alfvén wave driven unstable by the hot particle density gradient. These modes can be unstable even when the injection speed is sub-Alfvénic. The radial losses of beam ions are of comparable magnitude for both perpendicular and parallel injection. This is contrary to theoretical prediction, which says that the losses during tangential injection should be noticeably smaller [4]. The analysis of these global modes may also be important for the tangential beam injection problem in ITER and evaluation of the radial transport of energetic ions, and the ultimate current drive efficiency achievable.

Because of the documentation of experimental existence of these global modes, and the fact that these are thought to be the most dangerous, the study of *global* Alfvén modes driven unstable by the beams may become part of the study plan. Whether these modes should be the ones to concentrate on in the stability study should be a subject brought up at an appropriate physics co-ordinating meeting.

The numerical study of the WKB modes would involve calculating the ion distribution function driven away from thermodynamic equilibrium by: 1) near parallel injection of energetic ions and 2) the resulting Alfvén wave turbulence. The CQL code [2] would be useful for this purpose. From these distributions, the current drive efficiency could be computed, and

differences noted compared to the classical theory of Cordey [5]. Similar quasilinear distribution function calculations were performed by Berk and Stewart [6] including the ion cyclotron turbulence relevant to mirror machines.

The study of the global modes would require the calculation of MHD and kinetic toroidal eigenmodes, a task that several groups have worked on for the last 6 years or so. See for example, Refs. [7,8] and references within. The Fokker-Planck calculations would have to include a flux surface (radial) coordinate to treat the global nature of the problem. CQL can treat the required three dimensions ($p_\perp, p_\parallel, r$), but at cost of core space and execution time. The treatment of these modes will be much more involved than the WKB analysis.

The numerical work would eliminate the need to evaluate any approximate distribution function width because the matrix elements would be calculated with the complete distribution function directly in the code.

The analytic work would involve primarily the calculation of growth rates of the Alfvén modes using an analytically determined distribution function. The new aspect of this work compared to that previously reported [1] would be that the beams would be injected at an angle with respect to the magnetic field, the case relevant to ITER. Berk, *et. al.* [1] looked at off axis injection, but only transiently. The the analytic work may provide additional understanding of the underlying physics of steady-state non-tangential injection on the Alfvén wave problem.

Another potentially interesting semi-analytic calculation that could be done in parallel with the detailed Fokker-Planck calculations is to obtain the saturated quasi-linear distribution function using simplified forms for the particle-particle and wave-particle collision operators. The kinetic equation would be solved by the method of successive substitutions. The point where the iteration would be cut off would be when the growth rate at a given electric field strength converges to a single value. The convergence of this scheme is not assured for an arbitrary initial state, so there might have to be some care taken in this area. The self-consistent electric field strength would be determined in much the same way as planned for the detailed Fokker-Planck work; namely it would be varied until the damping on electrons and alphas just balance the growth of the mode. A simple $\langle v_\parallel \rangle$ moment will yield the desired current drive degradation factor.

The remainder of this memo outlines briefly relevant previous work in the microstability study, and discusses what we currently think the important modes are for ITER current drive, both local and *global* modes. The general results are not new, but this is the first place they have been evaluated for ITER parameters.

2 Background

When high-power neutral beam injection was first considered for tokamaks, Berk, et. al. [1] investigated the microstability of these two-component plasmas, in the approximation that the wavelength is short compared to characteristic spatial scalelengths (i. e. WKB). These authors examined a large variety of modes, and found the compressional and shear Alfvén waves were the most likely to be driven unstable, if the beam speed exceeded the Alfvén speed. Soon after this work, other authors examined different aspects of the Alfvén wave stability problem, including the non-linear saturation of the mode, and the effect of non-circular cross section on stability. A partial bibliography of this subsequent work may be found in the memo by Papanikolaou [9].

Recently there has been experimental evidence[3] of MHD bursting activity in small tokamaks (PBX) when beams are injected tangentially, resulting in radial losses of beam ions. Some of the mode's characteristics are similar to "fishbones" observed during perpendicular beam injection, which are thought to be a $m = 1$ long wavelength form of the shear Alfvén wave driven unstable by the hot particle density gradient. The analysis of these global modes may also be important for the tangential beam injection problem in ITER and evaluation of the radial transport of energetic ions. Within the last year, several authors [8,10] examined the destabilization of the global modes by alpha particles, which of course are also "injected" above the Alfvén speed, and have a radial pressure gradient. Depending on its perceived importance, the study of global Alfvén modes driven unstable by the beams may become part of the study plan.

3 Estimates for ITER

The free energy of the beam distribution which can drive the Alfvén waves unstable arise from parallel velocity space gradients, radial density gradients, and magnetic field curvature. The work of Rosenbluth and Rutherford [11] showed that the density gradient driven instabilities could be quasilinearly stabilized by a flattening of the beam radial profile near the center of the discharge. It is of interest to note that the beam density threshold for the density gradient driven instability is very low, but the stabilization mechanism was thought then strong enough to make this mode unimportant. The fishbone activity in present tokamaks suggest that the stabilization was not as strong as originally thought.

As a first look at the problem, however, we concentrate on the velocity space gradient driven modes.

Balancing the growth of the modes resulting from the free energy of the slowing down distribution with the damping provided by the electrons and alpha particles, a critical beam density relative to the electron density can be derived for both the compressional and shear modes.

One has the choice of evaluating the parameters related to the shape of the distribution function needed for the critical density and growth rate calculations with either an analytic form or from the results of a Fokker-Planck code [2]. We have obtained such a Fokker-Planck distribution for ITER. The contour plot of the distribution function in $(p_{\parallel}, p_{\perp})$ space is shown in Figure 1.

Since the code does not presently output the desired moments, and accurate estimates are important to properly calculating thresholds and growth rates, initial estimates will be made from the model distributions from Ref. [1]. The code will soon be modified to calculate these moments, so a more accurate distribution will be used during the study. Moreover, the critical density and growth rates are calculable directly by the code from moments of the quasilinear diffusion operator without approximation.

3.1 ITER Parameters

The parameters that we have assumed for ITER are shown in Table 1. We consider a radius close to the center because the local Alfvén speed is the

species 1 distribution function contour plot

Figure 1: Contour Plot of Ion Distribution in ITER

Parameter	Variable Name	Units	Value
Major Radius	R_0	cm	432.
Minor Radius	a	cm	142.
Radial Location	$\frac{r}{a}$	-	0.15
Plasma Elongation	κ	-	2.3
Toroidal Magnetic Field	B_0	T	4.95
Poloidal Magnetic Field	$B_P(a)$	T	1.11
Safety Factor at r	$q(r)$	-	1.5
Mean Plasma Density	$\bar{n}$	10^{13} cm^{-3}	7.2
Peak Electron Temperature	T_e	keV	23.6
Electron Thermal Speed	v_e	cm/sec	9.1×10^9
Peak Ion Temperature	T_i	keV	21.2
Ion Thermal Speed	v_i	cm/sec	1.42×10^8
Beam Injected Species	-	-	deuterium
Beam Injection Energy	E_{inj}	MeV	1.0
Parallel Beam Speed	$v_b \cos \theta_{inj}$	cm/sec	8.2×10^8
Beam Injection Angle	θ_{inj}	degrees	33.0
Ion Cyclotron Frequency	ω_{ci}	sec^{-1}	2.37×10^6
Central Alfvén Speed	$V_A(0)$	cm/sec	6.2×10^8
Beam-Electron Density Ratio	$\frac{n_b}{n_e}$	-	8.3×10^{-3}

Table 1: ITER parameters assumed for the stability calculation

lowest there, and that is where the most current is expected to be driven.

In toroidal geometry only certain values of the parallel wavevector, $k_{\parallel}$, are possible. This quantization condition can be expressed in terms of the toroidal connection length, qR_0 , and the toroidal and poloidal mode numbers, n and m respectively, by:

$$k_{\parallel} = \frac{1}{q(r)R_0} (m - nq(r)) \quad (1)$$

Typically, we look at long wavelength modes with m and n low integers. These modes usually have the fastest growth rates and lowest density thresholds. We shall take $m = 1$ and $n = 0$ for the calculations presented here. However, we note that to treat these long wavelengths properly the wave equations need to be solved for the radial mode structure, since the plasma parameters vary appreciably in one wavelength. Thus the results we present here based on the local approximation should be considered at best only qualitatively correct.

3.2 Compressional Alfvén Modes

The properties of the compressional waves will now be considered. For frequencies low enough so the condition $\frac{k_{\parallel}}{k_{\perp}} > \sqrt{\frac{\omega}{\omega_{ce}}}$, the wave electric field is polarized perpendicular to the plane containing the wave vector, $\hat{k}$, and the steady magnetic field, $\hat{B}$. This field polarization is accompanied by a perturbed parallel magnetic field, which provides the coupling to the ions through transit-time magnetic pumping. A simplified dispersion relation for the compressional wave is obtained from the work of Ross, *et. al.* [7]:

$$\omega = \frac{kV_A}{\sqrt{1 - \frac{11}{8}(k_{\perp}\rho_i)^2}} \quad (2)$$

Combining this with the condition for the wave phase velocity to match the beam particle parallel velocity, $\omega = k_{\parallel}v_b \cos \theta_{mj}$, one obtains the relationship between $k_{\parallel}$ and $k_{\perp}$, used here to calculate $k_{\perp}$ for a given k :

$$\left(\frac{k_{\perp}}{k_{\parallel}}\right)^2 = \frac{\left(\frac{v_b}{V_A} \cos \theta_{mj}\right)^2 - 1}{1 + \frac{11}{8}(\rho_i k_{\parallel})^2 \left(\frac{v_b}{V_A} \cos \theta_{mj}\right)^2} \quad (3)$$

Parameter	Variable Name	Units	Value
Parallel Wave Vector	$k_{\parallel}$	cm^{-1}	1.54×10^{-3}
Perpendicular Wave Vector	$k_{\perp}$	cm^{-1}	1.33×10^{-3}
Wave Frequency	ω	sec^{-1}	1.26×10^6
Critical Density Ratio	η_{crit}	-	3.13×10^{-4}
Growth Rate	γ	sec^{-1}	2000.

Table 2: Parameters related to the Wave Stability Calculation for the Compressional Mode

For the compressional modes, the microstability properties can be obtained directly from the infinite medium quasilinear theory. For these estimates, until the Fokker-Planck code is able to generate the proper quantities, we use the analytic distributions given in Reference [1]. Using the analytic distribution, the critical density is given by:

$$\eta_{crit} \equiv \frac{n_b}{n_e} = \frac{m_e v_e \alpha_0^2 \tau_f}{m_b v_b 6\sqrt{\pi}} \quad (4)$$

where n_b and n_e are the beam and electron densities, respectively, and the m 's and v 's are the masses and speeds. The parameter α_0 is related to the width in $v_{\perp}$ of the ion distribution, and the parameter $\tau_f = \frac{1}{3} \ln(1 + u_b^2)$ where $u_b = \frac{v_b}{v_f}$. The parameter v_f is the speed at which the contributions from electron drag and ion pitch angle scattering are equal, and is given by $v_f^2 = \frac{3\sqrt{2}}{4} \frac{m_e}{m_i} v_e^2 \hat{Z}$.

The growth rate of the instability, γ , evaluated for the analytic distribution is given by:

$$\gamma = \frac{3\pi\eta_{crit}k_{\perp}^2 k V_A m_b}{\alpha_0^2 \tau_f k_{\parallel}^2 m_i} \quad (5)$$

Table 3.2 shows the relevant wave parameters for the compressional Alfvén mode.

The parameter α_0 must be carefully chosen by examination of the distribution function and the particle source. For the case in Table 3.2, we take the width as $\alpha_0 \approx \frac{1}{\sin^2 \theta_{in}}$, appropriate for the $\perp$ width for off-tangential injection. This results in a very low threshold density compared to the beam density for ITER ($\times 25$ lower). For injection along the $v_{\parallel}$ axis, above the

energy where pitch angle scattering is important, the distribution width is more characteristic of the particle source. In that case, α_0 would be related to the width of the source distribution, as in Ref. [1]. Since the beams in ITER have quite low angular divergence, approaching a δ -function for all practical purposes, α_0 will be considerably larger ($\times 10^4$), and Eq. 4 would predict that the mode would not be excited. This counter-intuitive result for the growth arises because for the particles to interact with the compressional branch, there must be some perpendicular spread in the distribution. In a real plasma, the very steep perpendicular velocity gradients of the beam source would create some spread even above the critical energy.

The numerical work would eliminate the need to evaluate any approximate distribution function or source width because the matrix elements would be calculated with the complete distribution function directly in the code. This suggests that the numerical work should get underway as soon as practical.

3.3 Shear Alfvén Mode

For the shear mode, the electric field lies in the plane of the wavevector and magnetic field, nearly perpendicular to $\vec{B}$. For small parallel electric fields and infinite medium theory, there is no strong coupling between shear wave fields and distribution function phase space gradients (both $v_{\parallel}$ and r), in contrast to the compressional wave case discussed above. Therefore, Ref. [1] included an *ad. hoc.* curvature drift term to provide the coupling.

The dispersion relation for this mode can be obtained by using the relationships derived by Ross, et. al. [7]. Since $\frac{\omega}{k_{\parallel} v_A}$ is somewhat less than unity, the “warm electron” limit is appropriate. After using the phase velocity resonance condition, the expression for $k_{\perp}$ can be expressed:

$$k_{\perp}^2 = \left[\frac{3}{8} \left(\frac{v_b \cos \theta_{mj}}{V_A} \right)^2 \rho_i^2 + \rho_s^2 \right]^{-1} \left[\left(\frac{v_b \cos \theta_{mj}}{V_A} \right)^2 - 1 \right] \quad (6)$$

We examine the same $m = 1$, $n = 0$ mode as we did before, and get the same $k_{\parallel}$, but the resulting $k_{\perp}$ is much larger, suggesting that the shear mode has a much shorter wavelength than the compressional mode.

Parameter	Variable Name	Units	Value
Parallel Wave Vector	$k_{\parallel}$	cm^{-1}	1.54×10^{-3}
Perpendicular Wave Vector	$k_{\perp}$	cm^{-1}	1.07
Wave Frequency	ω	sec^{-1}	1.26×10^6
Beam Gyoradius	ρ_b	cm	4.1
Background Ion Gyoradius	ρ_i	cm	0.596
Sound Speed Gyoradius	ρ_s	cm	0.628
Distribution Velocity Spread	$\frac{\Delta v_b}{v_b}$	-	0.4
Critical Density Ratio	η_{crit}	-	3.23×10^{-3}
Growth Rate	γ	sec^{-1}	1.93×10^5

Table 3: Parameters related to the Wave Stability Calculation for the Shear Mode

The density threshold for the mode, from Ref. [1] is given by:

$$\eta_{crit} = \sqrt{\frac{m_e}{m_i}} \frac{m_i}{m_b} \frac{v_i}{v_b} \left(\frac{\Delta v_b}{v_b} \right)^2 (k_{\perp} \rho_b)^2 (k_{\parallel} R)^2 \quad (7)$$

The growth rate, also from that reference, is found to be:

$$\gamma = \eta_{crit} \frac{m_b}{m_i} \left(\frac{v_b}{\Delta v_b} \right)^2 \frac{1}{k_{\parallel} R} \frac{v_b}{R} \quad (8)$$

Notice that the critical density ratio has the opposite dependence on the distribution width (Δv_b here or $\frac{1}{\sqrt{\alpha_0}}$ for the compressional mode). This suggests that narrow distributions result in lower thresholds for this mode. This is more physically reasonable, since the narrower distributions around the injection point carry more free energy which can be tapped. For the shear case, the coupling depends on the zero-order Bessel function, $J_0(k_{\perp} \rho)$, so no perpendicular width is necessary for the coupling.

Table 3.3 shows parameters related to the shear mode evaluated for ITER. By examining Table 3.3, we can see that the shear mode also seems to be unstable for ITER. The threshold density is lower ($\times 2.6$) than the ITER beam density. The width of the distribution was estimated from the distribution depicted in Figure 1. This is a crude estimate, and real thresholds must wait for a more accurate calculation of the ion distribution

function. Although not as far from the critical density as the compressional mode, the growth rate is substantially larger, on the order of one-tenth the real part of the frequency.

4 Acknowledgements

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1 ITER Current Drive Efficiency in the Presence of Alfvén Wave Turbulence

1.1 General

The baseline for ITER uses high energy neutral beam injection to drive a steady-state current. Beam penetration and standard current drive efficiency considerations lead us to consider energies in the range of 1-3 MeV. Apart from the ambitious extrapolation from present source and accelerator technologies, there is also concern that the high energy beams with speeds above the Alfvén speed ($V_A = \frac{B}{\sqrt{4\pi m_i n_i}}$) will excite normal modes of the plasma which could, through enhanced slowing down of the beam, degrade the current drive efficiency. In ITER, the Alfvén energy near the plasma center is about 360 keV.

A model exists in the MUMAK code and the TETRA systems code to take into account the degradation of current drive efficiency by Alfvén waves. The model assumes that the beam particles instantaneously slow down to the Alfvén speed, resulting in a current drive efficiency which corresponds to injection at the Alfvén speed. Because the efficiency improves with increasing injection energy, the performance of ITER suffers when the model is employed. The model is thought to overestimate the effect. The complete depopulation of the region of velocity space above V_A would remove the drive for the instability. Velocity space gradients would be favorable to allow diffusion back into the depopulated region. The self-consistent solution would be one in which slowing down progresses only to the point when the instability drive vanishes. The actual degradation in current drive efficiency is best obtained by the formulation and solution of a two dimensional quasilinear diffusion problem.

We have already performed a cursory look at the density thresholds and growth rates for ITER using the results of Ref. [1], and have discovered that both the compressional and shear Alfvén modes could be important. These calculations did not address the quasilinear aspect of the problem in the sense that no relaxed distribution functions were calculated. It is the goal of the present work to calculate the quasilinearly flattened distribution function.

For the compressional mode, the perpendicular width of the distribution function is an important parameter in determining the degree of instability of the mode. For ITER, the minimum beam density for the compressional mode to become unstable is about a factor of 25 lower than the beam density in the baseline. The growth rate of the mode at the calculated beam density is approximately $\gamma \approx 2.3 \times 10^5 \text{ sec}^{-1}$. This growth rate is quite similar to the Fokker-Planck results reported in the next section.

The shear mode also appears to be unstable for ITER. For a beam distribution plateau width which is consistent with a Fokker-Planck calculation by the CQL code [2], the density threshold for this mode is a factor of 2.6 lower than the beam density for ITER. The growth rate at the calculated beam density is quite

large, on the order of one-tenth the real part of the frequency, $\gamma \approx 5.2 \times 10^5 \text{ sec}^{-1}$. We have yet to do a Fokker-Planck study for the shear mode.

Recently, there has been experimental evidence[3] of MHD bursting activity in small tokamaks (PBX) at low density when beams are injected tangentially, resulting in radial losses of beam ions. Some of the mode's characteristics are similar to "fishbones" observed during perpendicular beam injection, which are thought to be a $m = 1$, $n = 1$ long wavelength form of the shear Alfvén wave driven unstable by the hot particle density gradient. These modes can be unstable even when the injection speed is sub-Alfvénic. The radial losses of beam ions are of comparable magnitude for both perpendicular and parallel injection. This is contrary to theoretical prediction, which says that the losses during tangential injection should be noticeably smaller[4]. The analysis of these global modes may also be important for the tangential beam injection problem in ITER and evaluation of the radial transport of energetic ions, and the ultimate current drive efficiency achievable. Because of the confirmed existence of these global modes, and the fact that these are thought to be the most dangerous, the study of global Alfvén modes driven unstable by the beams may become part of the study plan.

1.2 The Local Quasilinear Calculation

The first step in calculating the effect of turbulence on neutral beam current drive is the examination of the WKB modes. We have developed a Simplified Fokker-Planck (SFP) code which solves for the beam distribution function. The model includes a quasilinear wave-particle interaction operator appropriate for both Alfvén wave polarizations. From this distribution, the current drive efficiency could be computed and compared to the classical theory of Cordey [5]. Similar quasilinear distribution function calculations were performed for mirror machines by Berk and Stewart [6] including the ion cyclotron turbulence relevant to that case.

The electric field strength of the turbulence is treated as a parameter in the code. The self-consistent electric field strength would be determined by varying it until the damping on electrons and alphas just balance the growth of the mode. A simple $\langle v_{\parallel} \rangle$ moment obtained from the relaxed distribution will yield the desired current drive degradation factor, when compared to the moment without the effect of turbulence.

Typically, we look at long wavelength modes with m and n low integers. These modes usually have the fastest growth rates and lowest density thresholds. We shall take $m = 1$ and $n = 0$ for the calculations presented here. However, we note that to treat these long wavelengths properly the wave equations need to be solved for the radial mode structure, since the plasma parameters vary appreciably in one wavelength. Thus, the results we present here based on the local approximation should be considered only approximately correct.

The compressional wave will now be considered. For frequencies low enough

Figure 1: The Collisional Ion Distribution Function in ITER

so the condition $\frac{k_A}{k_y} > \sqrt{\frac{\omega}{\omega_{ci}}}$, the wave electric field is polarized perpendicular to the plane containing the wave vector, $\vec{k}$, and the steady magnetic field, $\vec{B}$. This field polarization is accompanied by a perturbed parallel magnetic field, which provides the coupling to the ions through transit-time magnetic pumping.

Figure 1 shows a contour plot in normalized perpendicular and parallel velocity space of the ion distribution function in ITER. This is the distribution function without the effects of quasilinear diffusion. The 20 contours begin a factor of 420 below the $U = 0$ peak of the distribution function, and neighboring contours differ in value by a factor of 1.2. The injection point at 1.6 MeV is clearly present. The growth rate of the mode at this energy is $\gamma \approx 7 \times 10^4 \text{ sec}^{-1}$.

Figure 2 shows what happens when turbulence is allowed to relax the distribution function. The broadening of the distribution function near the injection point is a direct result of the diffusion caused by the Landau resonance $\omega - k_{\parallel} v_{\parallel}$. Figure 3 shows the parallel distribution function $f(v_{\parallel})$ for the relaxed distribution function. The slowing down of the beam injected ions occurs only in the parallel velocity range between the Alfvén speed and the parallel beam speed. Below V_A , the mode is evanescent, and above $V_{\parallel b}$ there would be no free energy drive to feed the turbulence.

We compare the current drive achieved with and without the turbulence present and find that the current drive efficiency is reduced by only about 10%. This degradation factor is considerably smaller reduction than obtained with the pessimistic model described previously. The small reduction in efficiency

Figure 2: The Ion Distribution Function Relaxed by Turbulence in ITER

due to the compressional mode can be understood by examining the velocity space integrands of the current drive and the rf power transfer. We have named these the current drive and rf "importance" functions, respectively. The volume under these surfaces is proportional to the current drive and rf power transfer to the unstable waves. Figures 4 and 5 display these importance functions for the collisional distribution. Notice that the regions where the current drive has its largest contribution is not where the rf interaction is the strongest. The effect of the quasilinear stabilization is to flatten the distribution the most in the region where the rf interaction is the strongest. Because this is at a different region in phase space than where the current drive has its largest contributions, a small effect is indeed expected.

We have yet to look at the shear branch of the Alfvén wave to assess its effect on current drive. This will be considered as an extension of the present work. There is reason to believe that the shear mode will be more important than the compressional mode for reducing the current drive efficiency. The reason for this can be understood by the $v_{\perp}$ dependence of the rf-particle interaction. For the compressional mode, the interaction scales as $v_{\perp} \times J_1^2(k_{\perp} \rho_i) \propto k_{\perp}^2 v_{\perp}^4$. This puts strong weighting to larger $v_{\perp}$ values. Since the beams are injected nearly parallel to the field, the rf-particle interaction is most important for $v_{\perp}$ larger than the perpendicular component of the injection velocity. The shear mode, on the other hand, has an interaction $v_{drift}^2 \times J_0^2(k_{\perp} \rho_i) \propto v_{\parallel}^2$ for small $k_{\perp}$ near $v_{\perp} \approx 0$. This may result in the dominant current carrying particles being effected strongly by the quasilinear flattening.

Figure 3: The Relaxed Parallel Ion Distribution Function in ITER

Figure 4: The Current Drive Importance Function in ITER

Figure 5: The rf Importance Function for ITER

1.3 Further Work

Continue work on the general role of Alfvén waves in the performance of ITER. The work would probably center around the following topics:

- Continue the assessment of the degradation of current drive efficiency by short wavelength Alfvén modes driven unstable by the presence of super-Alfvénic beam particles. The quasilinearly flattened ion distribution and wave momentum transfer to the electrons are currently being calculated, and will determine the modification to the current drive efficiency.
- Assess the impact of unstable global (as opposed to short wavelength) Alfvén modes on neutral beam current drive and hot particle confinement. We would examine the confinement of both hot injected species and hot alpha particles. Stabilization mechanisms will be sought.
- Work with the D-III-D experimentalists in defining and analyzing experiments to identify the Alfvén instability during neutral beam injection.

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I have developed a two-dimensional quasi-linear code in order to assess the degradation of neutral beam current drive efficiency due to turbulence excited by the injection of super-Alfvénic particles. My code uses Kennel-Englemann quasilinear theory for the compressional and shear Alfvén waves. The beam distribution function has been expanded in Legendre polynomials, resulting in a set of coupled ODE's in the speed coordinate. I have calculated growth rates of the modes using the collisional distribution, and I obtain the relaxed distribution for a given turbulence level. I obtain the self-consistent turbulence level by balancing the mode growth from beam particles and the mode damping from electrons and alpha particles. I will present quantitative results for the reduction in current drive efficiency.

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Illuminating discussions with T.K. Smeed and W.M. Nevins are gratefully acknowledged.

MOTIVATION

- The injection of energetic beam particles above the Alfvén speed can, if the beam density is high enough, excite a velocity- space microinstability. The excited turbulence has the polarization of the compressional and shear Alfvén branches.
- The turbulence can interact with the beam particles, and for nearly parallel injection, results in an enhanced slowing down of the beam.
- For ETR's and reactors which use neutral beams for steady-state current drive, these modes must either be avoided or the reduction in beam density must be accounted for in a calculation of current drive efficiency.

CODE DESCRIPTION

- Develop a code (SFP) which addresses the specific nature of the quasilinear relaxation of the beam distribution function. Gives flexibility in making code improvements.
- Two-dimensional ($v - \cos \theta$) Fokker-Planck, with particle-particle and wave-particle interactions. Particle source is Gaussian with source strength, energy and angle widths treated as parameters.
- Mode coupling is ignored.
- Local approximation implicit. Global radial structure for shear mode must be considered in a separate study.
- Modes driven by hot particle density gradients are to be considered in future work.
- Model turbulence spectrum shapes assumed. Similar to shapes obtained by Kulygin, et al (1976).

BASIC EQUATIONS

- The two dimensional Quasilinear equations can be written schematically as:

$$\frac{\partial f_b(\mathbf{v}, t)}{\partial t} = C_p(f_b) + C_q(f_b) + S(\mathbf{v}), \quad (1)$$

and the turbulent energy density obeys:

$$\frac{\partial W(\mathbf{v}, t)}{\partial t} = 2(\gamma_b + \gamma_e + \gamma_a) W(\mathbf{v}, t). \quad (2)$$

- These equations are solved in the steady state to determine the saturated quasilinear state.

PARTICLE COLLISION OPERATOR

- The particle collision operator is linearized, and is valid in the speed range $1.5v_{th,i} \leq v \ll v_{th,e}$. We follow Cordey and Core (1974), the operator has the Legendre pitch angle operator, and is second order in the speed co-ordinate.

WAVE POLARIZATIONS, AND DISPERSION RELATIONS

- The parallel wave vector is determined by:

$$k_{\parallel} = \frac{1}{q(r)R_0} (m - nq(r)) \quad (3)$$

- The wave frequency, ω , is determined by:

$$\omega = k_{\parallel} v_{\parallel,b} \quad (4)$$

• Compressional Mode:

Polarized perpendicular to $\mathbf{B}$ and $\mathbf{k}_\perp$.

Drive from "inverse" transit time magnetic pumping.

$$\omega = \frac{k V_A}{\sqrt{1 - \frac{11}{8} (k_\perp \rho_i)^2}} \quad (5)$$

When the phase velocity resonance condition $\frac{\omega}{k_\parallel} \approx v_\parallel$ is used, $k_\perp$ can be written:

$$\left(\frac{k_\perp}{k_\parallel}\right)^2 = \frac{\left(\frac{v_\parallel}{V_A}\right)^2 - 1}{1 + \frac{11}{8} (\rho_i k_\parallel)^2 \left(\frac{v_\parallel}{V_A}\right)^2} \quad (6)$$

• Shear Mode:

Polarized along $\mathbf{k}_\perp$, perpendicular to $\mathbf{B}$.

Drive is from parallel velocity gradients coupling to gradient and curvature drifts.

$$k_\perp^2 = \left[\frac{3}{8} \left(\frac{\omega}{V_A}\right)^2 \rho_i^2 + k_\parallel^2 \rho_s^2 \right]^{-1} \left[\left(\frac{\omega}{V_A}\right)^2 - k_\parallel^2 \right] \quad (7)$$

After using the parallel phase velocity resonance condition

$\frac{\omega}{k_\parallel} \approx v_\parallel$ is used, $k_\perp$ can be written:

$$k_\perp^2 = \left[\frac{3}{8} \left(\frac{v_\parallel}{V_A}\right)^2 \rho_i^2 + \rho_s^2 \right]^{-1} \left[\left(\frac{v_\parallel}{V_A}\right)^2 - 1 \right] \quad (8)$$

WAVE PARTICLE INTERACTION OPERATOR

- The quasilinear operator can be written:

$$C_{ql}(f_b) = \frac{\partial}{\partial v_{\parallel}} \left(\mathcal{D}_{ql} \frac{\partial}{\partial v_{\parallel}} f_b \right) \quad (9)$$

- The diffusion coefficient $\mathcal{D}_{ql}$ is written for the compressional mode:

$$\mathcal{D}_{ql} = \frac{\pi Z^2 e^2}{m_b^2} \left(\frac{v_{\perp}}{v_{\parallel}} \right)^2 \frac{S(v_{\parallel})}{v_{\parallel}} |E|^2 J_1^2(k_{\perp}(k_{\parallel})\rho_b) \quad (10)$$

- The diffusion coefficient $\mathcal{D}_{ql}$ is written for the shear mode:

$$\mathcal{D}_{ql} = \frac{\pi Z^2 e^2}{m_b^2} \left(\frac{v_{\perp}}{v_{\parallel}} \right)^2 \frac{S(v_{\parallel})}{v_{\parallel}} |E|^2 \left(\frac{v_{\perp}}{v_{\parallel}} \right)^2 J_0^2(k_{\perp}(k_{\parallel})\rho_b) \quad (11)$$

- The function $S(v_{\parallel})$ is the spectrum shape, and presently is treated as a model.

PARAMETERS FOR ITER

Parameter	Variable Name	Units	Value
Major Radius	R_0	cm	432.
Minor Radius	a	cm	142.
Radial Location	$\frac{r}{a}$	-	0.15
Plasma Elongation	κ	-	2.3
Toroidal Magnetic Field	B_0	T	4.95
Poloidal Magnetic Field	$B_P(a)$	T	1.11
Safety Factor at r	$q(r)$	-	1.5
Central Electron Density	n_e	10^{14} cm^{-3}	1.4
Effective Charge State	Z_{eff}	-	1.5
Peak Electron Temperature	T_e	keV	23.6
Beam Injected Species	-	-	deuterium
Beam Injection Energy	E_{inj}	MeV	1.6
Beam Angular Spread	$\Delta\mu$	-	0.025
Beam Energy Spread	ΔE	keV	16
Beam Particle Source	S_0	$10^{12} \text{ cm}^{-3}\text{sec}^{-1}$	1.4
Beam Density	n_b	10^{12} cm^{-3}	1.8
Beam Speed to Critical Speed	$\frac{v_b}{v_c}$	-	1.9
Beam Injection Angle	θ_{inj}	degrees	33.0
Ion Cyclotron Frequency	ω_{ci}	sec^{-1}	2.37×10^8
Central Alfvén Speed	$V_A(0)$	cm/sec	6.2×10^8
Central Alfvén Energy	$E_A(0)$	keV	358

COLLISIONAL BEAM DISTRIBUTION IN ITER

SAMPLE COMPRESSIONAL
 MODE STABILITY RESULTS FOR
 ITER
 COLLISIONAL DISTRIBUTION
 ONLY

Parameter	Variable Name	Units	Value
Parallel Wave Vector	$k_{\parallel}$	cm^{-1}	1.92×10^{-3}
Perpendicular Wave Vector	$k_{\perp}$	cm^{-1}	2.4×10^{-3}
Wave Frequency	ω	sec^{-1}	2×10^6
Growth Rate	γ_b	10^5 sec^{-1}	2.65
Damping on Electrons	γ_e	10^4 sec^{-1}	-1.07
Damping on Alphas	γ_{α}	10^2 sec^{-1}	-2.5

BEAM DISTRIBUTION RELAXED BY COMPRESSIONAL MODE TURBULENCE

EXAMPLE SHEAR MODE STABILITY RESULTS FOR ITER COLLISIONAL AND SATURATED DISTRIBUTION

Parameter	Variable Name	Units	Value
Parallel Wave Vector	$k_{\parallel}$	cm^{-1}	1.54×10^{-3}
Perpendicular Wave Vector	$k_{\perp}$	cm^{-1}	1.82
Wave Frequency	ω	sec^{-1}	2×10^6
Beam Gyroradius	ρ_b	cm	5.22
Background Ion Gyroradius	ρ_i	cm	0.567
Sound Speed Gyroradius	ρ_s	cm	0.401
Growth Rate	γ_b	10^4 sec^{-1}	9.24
Damping on Electrons	γ_e	10^4 sec^{-1}	-3.66
Damping on Alphas	γ_{α}	10^2 sec^{-1}	-8.28
Wave Energy Density	$\frac{W}{nkT}$	10^{-6}	3
Collisional Time	τ_{coll}	sec	1.34
Rf Diffusion Time	τ_{ql}	sec	1.17
Beam Density	n_b	10^{11} cm^{-3}	9
C.D. Reduction	$\frac{\eta_{nb,qi}}{\eta_{nb,0}}$	-	0.5

BEAM DISTRIBUTION RELAXED BY SHEAR MODE TURBULENCE

THE STABILITY PROPERTIES OF ITER CAN BE IMPROVED BY JUDICIOUS CHOICE OF OPERATING PARAMETERS

- Broadening the source improves the stability properties of ITER.
- Changing the electron temperature changes plasma beta, the slowing down time, and the electron damping. Will impact the beam density and stabilizing influences.
- Changing the background density changes plasma beta, the Alfvén speed, beam trapping, and slowing down time. Alfvén speed effects wave-vector properties, and hence damping and growth.
- Changing the beam energy changes the beam trapping, wave- vector properties, and slowing down of the beam.
- Changing particle source strength changes beam density, and the drive for instability.
- All these changes must be consistent with plasma power balance.

Impact of Source Parameters on Compressional Mode Growth Rate

EXPERIMENTS ARE NEEDED TO ASSESS THE IMPORTANCE OF THE ALFVÉN MODES ON CURRENT DRIVE

- Experiments on D-III-D with high power injection are possible. MHD activity probes and charge exchange analysis could yield important information.
- Experiments on T-11 have been performed, and no evidence of instability was found. These experiments were performed under a set of conditions which this analysis would predict a plasma stable to the compressional Alfvén modes.

ASSUMED PARAMETERS FOR T-11

Parameter	Variable Name	Units	Value
Major Radius	R_0	cm	70
Minor Radius	a	cm	16
Radial Location	$\frac{r}{a}$	-	0.15
Plasma Elongation	κ	-	1.0
Toroidal Magnetic Field	B_0	T	0.87
Poloidal Magnetic Field	$B_P(a)$	T	0.13
Safety Factor at r	$q(r)$	-	1.5
Central Electron Density	n_e	10^{14} cm^{-3}	0.8
Effective Charge State	Z_{eff}	-	1.0
Peak Electron Temperature	T_e	keV	0.5
Beam Injected Species	-	-	hydrogen
Beam Injection Energy	E_{inj}	keV	22
Beam Angular Spread	$\Delta\mu$	-	0.1
Beam Energy Spread	ΔE	keV	2.2
Beam Particle Source	S_0	$10^{14} \text{ cm}^{-3} \text{ sec}^{-1}$	1.5
Beam Density	n_b	10^{11} cm^{-3}	5.69
Beam Speed to Critical Speed	$\frac{v_b}{v_c}$	-	1.72
Beam Injection Angle	θ_{inj}	degrees	26.0
Ion Cyclotron Frequency	ω_{ci}	sec^{-1}	8.3×10^7
Central Alfvén Speed	$V_A(0)$	cm/sec	1.5×10^8
Central Alfvén Energy	$E_A(0)$	keV	11.8

SAMPLE COMPRESSIONAL MODE STABILITY RESULTS FOR T-11

Parameter	Variable Name	Units	Value
Parallel Wave Vector	$k_{ }$	cm^{-1}	1.19×10^{-2}
Perpendicular Wave Vector	$k_{\perp}$	cm^{-1}	8.54×10^{-3}
Wave Frequency	ω	sec^{-1}	2.2×10^6
Growth Rate	γ_b	10^3 sec^{-1}	2.1
Damping on Electrons	γ_e	10^3 sec^{-1}	- 3.84

EXAMPLE SHEAR MODE STABILITY RESULTS FOR T-11

Parameter	Variable Name	Units	Value
Parallel Wave Vector	$k_{ }$	cm^{-1}	1.19×10^{-2}
Perpendicular Wave Vector	$k_{\perp}$	cm^{-1}	2.64
Wave Frequency	ω	sec^{-1}	2.2×10^6
Beam Gyoradius	ρ_b	cm	1.08
Background Ion Gyoradius	ρ_i	cm	0.262
Sound Speed Gyoradius	ρ_s	cm	0.186
Growth Rate	γ_b	10^4 sec^{-1}	5.58
Damping on Electrons	γ_e	10^4 sec^{-1}	-2.02

CONCLUSIONS AND RECOMENDATIONS

- Preliminary results indicate that the baseline ITER is unstable to both the shear and compressional Alfvén modes.
- Quasilinear relaxation of the distribution due to these waves reduces the current drive efficiency by a factor of 2.
- Careful choice of operating parameters can improve the stability picture.
- Results are sensitive to the details of the source of energetic particles.
- Experiments on D-III-D are very important in testing the theoretical predictions.
- Experiments on T-11 suggest that Alfvén waves won't be important.
- Code simulation of T-11 suggests the compressional mode is stable, and the shear mode unstable in that machine.
- Examination of the global shear mode is also important.

DRAFT

May 10, 1988

LLNL-ITER-88-042

MEMORANDUM

To: ITER Physics Distribution

From: R.B. Campbell

Subject: Neutral Beam Modeling of the T-11 Tokamak and Implications for Stability of Alfvén Waves in ITER.

1 Summary

Encouraging experimental results [1] have been obtained on T-11 under intense super-Alfvénic beam injection which suggest that Alfvén waves either do not exist or are benign.

Calculations performed with a two-dimensional quasilinear code I am developing suggest that:

- The compressional branch of the mode is stable in T-11.
- The shear branch is unstable, and the effect of the ensuing turbulence is to reduce the density of the hot ion plateau, *not* change the basic shape of the high energy tail.
- From my calculation, the presence of Alfvén instabilities in a device with injection of both super and sub-Alfvénic particles can be determined by comparing the plateau levels produced by each class of particle.
- The Alfvén waves deplete the plateau containing the super-Alfvénic particles, and leave the plateau levels from the slower particles largely unchanged. The T-11 data compared with the classical theory from [1]

normalized to the bumps produced by the sub-Alfvénic particles also demonstrates this principle.

- The primary effect observed is to reduce the density of ions above E_A , but does not eliminate neutral beam current drive from consideration for the steady-state ITER.
- A methodology for detecting the phenomenon is outlined in Section 7. Further experiments on large Tokamaks (e. g. D-III-D) is essential.

2 Introduction

Experimental results with neutral beam injection obtained from T-11 [1] suggest that despite dire theoretical predictions, the Alfvén instabilities do not exist, even though the full energy component of the beam exceeded the central Alfvén energy by a factor of two.

The experimental results were compared with a Fokker-Planck calculation which included only Coulomb collisions. The calculation was normalized to experiment at low energy ($\approx 1\text{keV}$), and agreement for the high energy tail distribution was very good. This lead the authors of Reference [1] to conclude that anomalous slowing down processes were not important in this plasma.

This experimental result is good news, and could add additional credibility to the proposal to use neutral beams injected above the Alfvén energy to drive a steady state current.

I believe it is worthwhile to apply additional theoretical analyses to the T-11 data and ascertain the likelihood that instabilities could have been present. The approach used here is to calculate the distribution function with and without the effects of slowing down due to Alfvén turbulence, and compare the characteristics of the two distribution functions. The additional slowing down will only act upon the super-Alfvénic species, and therefore relative changes in the plateau levels will be evident. Experimental measurement of the plateau levels and comparison with theory will then determine whether Alfvén instabilities are present.

3 Previous Work

The free energy of a beam distribution of sufficient density produced by the injection of particles above the Alfvén energy drives Alfvén waves unstable from parallel velocity space gradients and radial density gradients. The work of Berk, et. al. [2] examined a large variety of modes for a beam-driven Tokamak and concluded that unstable Alfvén modes could be quite serious. Parallel independent work by Kulygin, Mikhailovskii, et. al. [3] also concluded that the anomalous slowing down of the beam by turbulence was considerably shorter than the slowing down time from Coulomb collisions. The work of Rosenbluth and Rutherford [4] showed that the density gradient driven instabilities could be quasilinearly stabilized by a flattening of the beam radial profile near the center of the discharge. Most of the work in the mid 1970's suggested that there might be serious problems associated with the injection of super-Alfvénic particles.

I am developing a two-dimensional ($v_{\perp} - v_{\parallel}$) quasilinear code and have applied it to ITER [5]. Preliminary calculations indicate that both the shear and compressional branches are unstable for typical ITER conditions, and the quasilinearly saturated state possesses a current drive efficiency lower than that predicted from a theory including only Coulombic slowing down. I find that the shape of the hot tail is *not* modified significantly in the presence of turbulence and only the *level* of the plateau (i. e. beam density) is reduced. It is possible that a level change in the distribution could have been missed in previous T-11 code-experiment comparisons because of the normalization procedure.

Because of these results, it is imperative that experimental work be reviewed and considered, and makes the T-11 work particularly important. It also seemed reasonable to benchmark the code used to model ITER with experimental data.

4 Parameters for T-11

The purpose of this section is to present numerical values for the parameters used in the quasilinear calculation and to present the methodology employed in arriving at the input values needed which were not directly available. I have obtained parameters for the calculation directly from references [1] and [6]

when practical, and inferred others when the information was not explicitly given. The results of the calculation depend on the choice of effective beam energy spread, beam angular spread, and particle fractions present in the full, half, and third energy components. These parameters could not be obtained directly from the published literature that I am aware of. Whenever possible, values were chosen so as to be conservative in the stability estimates and quasilinear relaxation.

Background deuterium and electron densities and temperatures are directly obtainable from Refs. [1,6]. I took a small (1.5 %) oxygen impurity concentration in the deuterium background plasma. The electron density is $9.0 \times 10^{13} \text{ cm}^{-3}$ for the calculations, and the Alfvén energy, E_A , is 10.6 keV. The full energy component is super-Alfvénic by a factor of two, the half energy component is borderline, and not expected to be important in the stability picture.

Because of toroidal geometry, exactly parallel beam injection in all flux surfaces is not possible. The charge exchange diagnostic could sample neutrals produced at various radii and hence injection angles. I determined the value of the "average" injection angle to be 17.6° by noticing from Reference [1] that 22KeV beams produced a parallel tail which drops off precipitously at 20keV.

The particle fractions of the full, half, and third energy components were derived from molecular fractions out of similar sources at LBL (the raw data is courtesy of Bill Cropper). The H^+ , H_2^+ , and H_3^+ particle fractions are 0.65, 0.175, and 0.175, respectively. After accounting for molecular breakup and energy dependent neutralization efficiency, I calculated fractions in the full, half, and third energy components of 0.4, 0.24, and 0.36, respectively.

The energy and angular widths of the particle source depend on the characteristics of the source and acceleration stage of the injector, as well as the injection geometry. Since I did not have detailed information in these areas, I simply took values which were the narrowest I could use with the numerical scheme I chose to solve the Fokker-Planck equation. This should result in conservative results, since the narrower sources produce particle distributions with more free energy to drive instabilities. The energy width was taken to be $\Delta E \approx 0.1 \text{ keV}$ and $\Delta \cos \theta \approx 0.025$.

The total beam power used in the calculations was 0.6MW, assumed to contain power at all three energy components. This power, after assuming a plasma volume ($4 \times 10^5 \text{ cm}^3$) and the particle fractions above, enabled me

Parameter	Variable Name	Units	Value
Major Radius	R_0	cm	70
Minor Radius	a	cm	17
Toroidal Magnetic Field	B_0	T	0.87
Central Electron Density	n_e	10^{14} cm^{-3}	0.9
Impurity Species	-	-	oxygen
Impurity Concentration	$\frac{n_i}{n_D}$	-	0.015
Effective Charge State	Z_{eff}	-	1.75
Peak Electron Temperature	T_e	keV	0.5
Beam Injected Species	-	-	hydrogen
Background Species	-	-	deuterium
Full Injection Energy	E_{inj}	keV	22
Angular Spread (all comps)	$\Delta\mu$	-	0.025
Energy Spread (all comps)	ΔE	keV	0.1
Particle Source (full)	S_{of}	$10^{14} \text{ cm}^{-3}\text{sec}^{-1}$	2.65
Particle Source (half)	S_{oh}	$10^{14} \text{ cm}^{-3}\text{sec}^{-1}$	1.6
Particle Source (third)	S_{ot}	$10^{14} \text{ cm}^{-3}\text{sec}^{-1}$	2.4
Beam Injection Angle	θ_{inj}	degrees	17.6

Table 1: Input Variables for T-11 Quasilinear Calculation

to calculate the source strength in each of the beam components. Table 1 summarizes the assumed input parameters in the quasilinear calculation.

There are several key parameters which can be quickly calculated and characterize the beam-background plasma system. These parameters can be compared between different machines and conclusions reached about the applicability of using data from one machine to predict the performance of another.

The beam density relative to the electron density, $\frac{n_b}{n_e}$, gives a measure of the importance of beam growth compared to electron damping. It is possible that there is instability drive present, but the hot ion density is not large enough for the mode to grow. The ratio of each beam speed to the critical speed (where the effects of electron drag and ion pitch angle scattering are comparable), $\frac{v_b}{v_c}$, gives information about how readily unstable, anisotropic distributions can be produced. The ratio of each beam energy to the Alfvén energy $\frac{E_b}{E_A}$ indicates to what degree the system is susceptible to the modes.

Parameter	Variable Name	Units	T-11	ITER
Central Electron Density	n_e	10^{14} cm^{-3}	0.9	1.4
Density Ratio	$\frac{n_b}{n_e}$	10^{-2}	1.54	1.28
Effective Charge State	Z_{eff}	-	1.75	1.5
Peak Electron Temperature	T_e	keV	0.5	23.0
Beam Injected Species	-	-	hydrogen	deuterium
Background Species	-	-	deuterium	50-50 D-T
Full Injection Energy	E_{inj}	keV	22	1600
Angular Spread (all comps)	$\Delta\mu$	-	0.025	0.025
Energy Spread (all comps)	ΔE	keV	0.1	16.0
Particle Source (full)	S_{of}	$10^{14} \text{ cm}^{-3}\text{sec}^{-1}$	2.65	0.014
Particle Source (half)	S_{oh}	$10^{14} \text{ cm}^{-3}\text{sec}^{-1}$	1.6	-
Particle Source (third)	S_{ox}	$10^{14} \text{ cm}^{-3}\text{sec}^{-1}$	2.4	-
Beam Speed to Critical Speed	$\frac{v_b}{v_e}$	-	1.72	1.9
Beam Collision Time	τ_{coll}	msec	5.5	1830
Beam Injection Angle	θ_{inj}	degrees	17.6	33.0
Central Alfvén Energy	$E_A(0)$	keV	10.5	358.0
Beam to Alfvén Ratio	$\frac{E_b(0)}{E_A(0)}$	-	2.1	4.5

Table 2: Comparison of Beam Parameters between T-11 and ITER.

The beam background collision time, τ_{coll} , is a measure of how efficiently a given source of particles can produce a hot particle density. Plasmas with low electron temperatures are less efficient in producing hot ion density than are those with higher T_e , all other factors remaining the same. Table 2 presents some important parameters (both input and calculated) for T-11 and compares them with ITER. The calculations are described in more detail in Section 5 below.

Comparing the T-11 parameters to those for ITER, we see that several of the dimensionless parameters are comparable. The most notable exception is the ratio of E_b to E_A . The ITER case appears to be deeper in the regime where an Alfvén instability could be observed. It may therefore be desirable to perform such experiments in a machine which could generate a large value of this ratio (such as D-III-D).

The beam-background collision time for T-11 is only 5 msec, due to the low electron temperature in that device. The beam density in both machines

is about the same. Because τ_{coll} is so short in T-11 compared to ITER, T-11 requires a considerably stronger particle source to produce the same hot density.

5 Fokker-Planck Calculations for T-11

The Fokker-Planck code which I am currently developing concentrates on the velocity space gradient driven modes. I am in the early stages of thinking about a radial ($r - v_{||}$) kinetic code which will address the global Alfvén mode.

Balancing the growth of the modes resulting from the free energy of the slowing down distribution with the damping provided by the electrons and alpha particles (if present), a critical beam density relative to the electron density can be derived for both the compressional and shear modes.

We consider a radius close to the center because the local Alfvén speed is the lowest there, and that is where observation of an instability is most likely.

Typically, we look at long wavelength modes with m and n low integers. We shall take $m = 1$ and $n = 0$ for the calculations presented here. However, we note that to treat these long wavelengths properly the wave equations need to be solved for the radial mode structure, since the plasma parameters vary appreciably in one wavelength. Thus the results we present here based on the local approximation should be considered suggestive of what the more detailed non-local calculation will produce.

The code solves numerically the quasilinear equations:

$$\frac{\partial f_b(\mathbf{v}, t)}{\partial t} = C_p(f_b) + C_\alpha(f_b) + S(\mathbf{v}), \quad (1)$$

and, the wave energy density obeys:

$$\frac{\partial W(\mathbf{v}, t)}{\partial t} = 2(\gamma_b + \gamma_e + \gamma_\alpha) W(\mathbf{v}, t). \quad (2)$$

These equations are solved in the steady state, so only the sinusoid ($\approx 6 \times \tau_{coll}$) curves in Ref. [1] can be modeled. The wave energy density equation is presently solved globally, in that we adjust the amplitude of the turbulence so that the total damping from electrons just balances the total growth from

the beam drive. The shape of the turbulence spectrum in $v_{\parallel}$ is treated as a model, and I have taken a conservative stance and assumed it to be a square function, non-zero only for values of $v_{\parallel}$ where there is strong drive. I am currently modifying the code to calculate the spectral energy density $W(v_{\parallel})$ self-consistently.

The particle collision operator, $C_p(f_b)$, I use is linearized and includes slowing down, energy dispersion, pitch angle scattering, and is valid down to ion speeds $\approx 1.5 \times v_{th,i}$.

The sources for full, half, and third energy components are Gaussian in speed v and $\mu = \cos \theta$, and have the form given by:

$$S_j(v, \mu) = A_j \exp \left[- \left(\frac{v - v_{b,j}}{\Delta v_{b,j}} \right)^2 - \left(\frac{\mu - \mu_{b,j}}{\Delta \mu_{b,j}} \right)^2 \right]. \quad (3)$$

The source normalization constant, A_j , is chosen to yield the volumetric source strength when integrated over velocity space.

The quasilinear operator is a diffusion operator in the parallel velocity coordinate. It's generic form is the same for both polarizations, and is given by:

$$C_q(f_b) = \frac{\partial}{\partial v_{\parallel}} \left(\mathcal{D}_q \frac{\partial}{\partial v_{\parallel}} f_b \right) \quad (4)$$

The diffusion coefficient, $\mathcal{D}_q$, for the compressional and shear modes will be given later in Sections 6.1 and 6.2.

5.1 Collisional Distribution

I looked at the T-11 problem first without including the effect of the turbulence on the distribution function (i. e. when equation 4 is not included). Figure 1 shows a contour plot of the ion distribution function in T-11 for this case. The perpendicular and parallel velocity components are normalized by the critical speed, v_c . Because the background temperatures are so much lower than the beam energy, I do not plot contours corresponding to energies below a value well below E_{inj} . Figures 2 and 3 show $f(E_{\parallel})$ and $f(E_{\perp})$ as a function of $E_{\parallel}$ and $E_{\perp}$ respectively. The distributions have been normalized to unity at 1 keV. There is nothing particularly suprising about these results, the code without the quasilinear effects has been compared to analytic results (see, for example [2]) and values agree in the appropriate limits.

Figure 1: Contours of Ion Distribution Function in T-11

Figure 2: Collisional Parallel Distribution versus $E_{||}$ in T-11

Figure 3: Collisional Perpendicular Distribution versus $E_{\perp}$ in T-11

6 Stability and Quasilinear Calculations

I will now proceed to assess the stability of the two branches of the Alfvén wave for T-11, and determine the possible impact on the ion distribution function.

The spectrum shape function, $W(v_{\parallel})$, is presently treated as a model. When I calculate stability of the collisional distribution, I use a square function which localizes the turbulence in $v_{\parallel}$ just below the beam injection point. This function spans a $v_{\parallel}$ range containing about 10 grid points. This spectrum amounts to the numerical equivalent of a delta function commonly used in analytic studies of micro-stability. When a quasilinearly saturated distribution is calculated, this spectrum is widened to model the effect of the instability drive becoming spread out away from the injection point as the distribution function flattens. The results can depend on the shape of this spectrum. An important extension of this work in progress is to calculate the spectrum shape based on the balance between beam drive and electron damping.

I perform a numerical stability analysis by calculating the rf-power input to the waves from the hot ions, using the collisional distribution, and comparing it to the rf-power transferred to the electrons from the unstable waves. The electron damping was computed analytically using a Maxwellian and a delta function spectrum. This calculation is done in the limit of zero turbulence level. If the mode is indeed unstable, I increase the turbulence level until the ion distribution is sufficiently flattened that the mode growth is just balanced by the damping. Comparing the distribution function and its moments (particularly density and parallel current) with and without the quasilinear diffusion some consistent assessment of the effects of turbulence on current drive efficiency can be made.

6.1 Compressional Alfvén Modes

For the compressional modes, the rf-interaction operator can be obtained directly from the infinite medium quasilinear theory. The diffusion coefficient, D_{qi} , is given by:

$$D_{qi} = \frac{\pi Z^2 e^2}{m_b^2} \left(\frac{v_{\perp}}{v_{\parallel}} \right)^2 \frac{S(v_{\parallel})}{v_{\parallel}} |E|^2 J_1^2(k_{\perp}(k_{\parallel})\rho_b) \quad (5)$$

The parameter $\rho_b = \frac{v_{\perp}}{\omega_{ci}}$ is the ion gyroradius, and J_1 is a Bessel function. The perpendicular wave vector, $k_{\perp}$, is a function of $k_{\parallel}$ through use of the dispersion relation, and hence a function of $v_{\parallel}$ by the phase velocity resonance condition $k_{\parallel} = \frac{\omega}{v_{\parallel}}$. The same is true for the shear mode as described in the next section.

The T-11 plasma was found to be at marginal stability, $\gamma_b + \gamma_e \approx 0$ for the compressional mode. I computed the growth rate with the code and found it to be $6.32 \times 10^3 \text{ sec}^{-1}$, compared to an electron damping rate of $-6.72 \times 10^3 \text{ sec}^{-1}$. An analytic estimate of γ_b from the formulas presented in Ref. [2] gives a value $6.82 \times 10^3 \text{ sec}^{-1}$. It therefore appears that the compressional mode is not a significant factor in the T-11 stability picture.

6.2 Shear Alfvén Mode

For the shear mode, the electric field lies in the plane of the wavevector and magnetic field, nearly perpendicular to $\vec{B}$. For small parallel electric fields and infinite medium theory, there is no strong coupling between shear wave fields and distribution function phase space gradients (both $v_{\parallel}$ and r), in contrast to the compressional wave case discussed above. Therefore, Ref. [2] included an *ad hoc* drift term, $v_D \cdot E$, to provide the coupling. For this case, the diffusion coefficient is given by:

$$\mathcal{D}_{\text{eff}} = \frac{\pi Z^2 e^2}{m_b^2} \left(\frac{v_{\perp}}{v_{\parallel}} \right)^2 \frac{S(v_{\parallel})}{v_{\parallel}} |E|^2 \left(\frac{v_D}{v_{\perp}} \right)^2 J_0^2(k_{\perp} \rho_b) \quad (6)$$

With all the same meanings as Eq. 5. The drift velocity includes both the magnetic drift terms due to gradients and curvature. The drift velocity is given by:

$$v_D = \frac{\frac{1}{2} (v_{\perp}^2 + 2v_{\parallel}^2)}{\omega_{ci} R_T} \quad (7)$$

The terms which describe power transfer from ions to waves and from waves to electrons are calculated using the shear mode quasilinear operator applied to the collisional distribution. I find that the shear mode is unstable, the growth rate for this mode exceeds the electron damping rate by a factor of 10. I calculate the growth rate to be $2.1 \times 10^6 \text{ sec}^{-1}$, slightly lower than the corresponding analytic estimate of $2.6 \times 10^6 \text{ sec}^{-1}$.

Notice that the parallel distribution function which yields this growth rate (see Figure 2) is essentially flat from the beam injection point down to energies approaching the critical energy. The simple notion of Landau damping/growth for electrostatic waves would suggest that this distribution would be essentially stable, since $\frac{df_{\parallel}(v_{\parallel})}{dv_{\parallel}} \approx 0$. The essential idea here is that the interaction of the beam particle with the wave for a given $v_{\parallel}$ is an average of $C_{\parallel}$, which gives more emphasis to regions of velocity space near the injection point. The positive slope in $v_{\parallel}$ of $f(v_{\perp}, v_{\parallel})$ is quite pronounced near the injection point. The weighting function can be inferred for each mode from the diffusion coefficient given by equations 5 and 6.

Now that I have shown that the code predicts instability using the collisional distribution, I want to determine the steady state where just as much power is being fed into the waves as is being damped on the electrons. On Figure 4 I show the parallel distribution function for the saturated state.

Notice that a change from a neutral $\frac{df_{\parallel}(v_{\parallel})}{dv_{\parallel}}$ to a positive one as shown in this figure does not necessarily mean that the distribution is more unstable. As described in the previous section, it is a *weighted* average of f which determines the stability. Figure 5 shows the distortion of the distribution under the influence of the turbulence. Comparing the spacing between the contours (each a factor of 1.2 apart) near the injection point to those in Fig. 1, we see that flattening has indeed occurred.

Comparing $f_{\parallel}(v_{\parallel})$ in Fig. 4 to Fig. 2, notice that the level of the plateau produced by the super-Alfvénic component is depleted due to slowing down by the Alfvén waves. The levels of the half and third energy plateaus decrease slightly, since the distribution function must connect smoothly between the regions above and below E_A . Notice also that the slope of the distribution just below $E_A = 10.8 \text{ KeV}$ is larger for the quasilinearly stabilized distribution. This is a natural consequence of continuity of slowing down current at an interface where the diffusion coefficient changes from purely collisional to a larger one containing quasilinear effects.

I calculate that the total beam density (due to full, half, and third energy components) is reduced to $9.1 \times 10^{11} \text{ cm}^{-3}$, a reduction of about a factor of 1.5 compared to the result for the collisional distribution.

From density moments of the collision and quasilinear operators over the v -space region occupied by the hot particles, I have computed the total slowing down time to be 1.3 msec, which is a combination of the time due

Figure 4: Stable Parallel Distribution versus $E_{\parallel}$ in T-11

Figure 5: Contours of Stable Distribution in T-11

to 1.7 Coulombic collisions of 1.9 msec, and the turbulence induced slowing down of 4.2 msec. For comparison, the purely collisional case yields a value of 2 msec. As a point of interest, I have computed these numbers for ITER including Alfvén waves, and the contributions to the slowing down are about equal.

7 Methodology for Detecting Alfvén Instabilities

My result for the shear mode suggests a way in which the presence of instabilities could be detected in an experiment where particles both above and below the Alfvén energy are injected. I would first perform a calculation intended to model the experimental charge exchange spectrum diagnostic without the effect of turbulence. I did not have enough specifics about the diagnostic on T-11 to do this here. Then, I would normalize my calculation to one of the peaks in the data corresponding to sub-Alfvénic injection. The reason for choosing this normalization is two-fold. First, these peaks should not be substantially modified by the turbulence and will clearly illustrate any reduction in the high energy tail. Second, the energy for normalization is high enough that low energy edge neutrals would not complicate the interpretation, as discussed in [6]. A direct comparison of the super-Alfvénic plateaus could then be made.

The T-11 theory-experiment comparison made in [1] can be used to illustrate this point. Figure 6 shows the T-11 data for the parallel charge exchange diagnostic with their calculation superimposed. This figure differs from the one in Ref. [1] in that I have renormalized the calculation to the lowest energy peak of the data. The high energy plateau is indeed depleted, perhaps due to Alfvén wave induced turbulence. The exact reduction is hard to read off the semilogarithmic plot, but a factor of 1.5 appears reasonable.

8 Conclusions

In this memo, I have shown through my own calculations and by recasting previous studies that Alfvén waves may have been present in T-11. The primary effect observed is to reduce the density of ions above E_A , but does

Figure 6: T-11 Experiment Theory Comparison, Renormalized to the Lowest Energy Beam Peak

not eliminate neutral beam current drive from consideration for the steady-state ITER.

I believe it is necessary for analysis and experiments to continue. More study and understanding of T-11 data could be very productive. Experiments on D-III-D should also be performed, incorporating some the methodology presented here. Because the Alfvén wave issue is important to the success of a neutral beam driven steady-state ITER, more effort from a number of groups should be expended.

9 Acknowledgements

I would like to acknowledge valuable discussions with W.M. Nevins and R.S. DeVoto.

References

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**Alfvén Wave Instabilities
in
Beam-Driven Tokamaks**

Robert B. Campbell (TRW/LLNL)

**Presented at Current Drive and Heating Workshop
Garching, FRG**

13-17 June 1988

ESSENCE OF TALK

- Models which assume that the beam slowing down due to Alfvén instabilities is instantaneous are quite pessimistic. Quasilinear calculations show that slowing down times due to Coulomb collisions and wave-particle interactions are on the same order, in the steady state.
- Preliminary results from a quasilinear calculation suggest fast ion current could be reduced by 25% due to Alfvén instabilities in the US/ITER proposed design ($R = 4.2m$, $E_{inj,max} = 1.6MeV$).

ESSENCE OF TALK (CONT)

- The stability picture for the shear mode can be improved if the apparent angular width of the source (and hence the ion distribution) is increased. The drive for instability is a velocity space integral of:

$$V^2 J_n^2 \frac{\partial f_b}{\partial v_{\parallel}^2} \approx J_n^2 (1 - \mu^2) \frac{f_b}{\Delta \mu^2},$$

where $\Delta \mu = \Delta \cos \theta$ is a measure of the beam angular spread. Two “knobs” which determine $\Delta \mu$ and can be controlled in the beam design are:

1. The distribution of beamlines across the plasma.
 2. The dimensions of beam footprint.
- Experiments on large tokamaks (e. g. D-III-D) are essential in studying the problem further.

PROBLEM

- The injection of particles with speeds along $\vec{B}$ larger than the Alfvén speed, $V_A = \frac{B}{\sqrt{4\pi\rho_i}}$, may introduce velocity space instabilities, if the ratio of hot density to electron density exceeds a critical value.
- These instabilities will tend to decrease the slowing down time of beam particles, hence reduce the current-carrying beam density. A reduction in beam density translates to a reduction in current drive efficiency.
- The worst possible case is that the injected particles slow down instantaneously to the Alfvén speed, depopulating velocity space above V_A .
- The real reduction in current drive efficiency will be less, because the quasilinear evolution will proceed only to marginal stability, not to complete depopulation.
- A two-dimensional ($v_\perp - v_\parallel$) quasilinear calculation is necessary for a quantitative assessment of the effect.

IMPLICATIONS FOR ITER

- The beam energy, E_b , for the steady-state US/ITER proposed design is 5.0 times the Alfvén energy, $E_A = \frac{1}{2}m_b V_A^2 \approx 300\text{keV}$, and the beam density, $n_b \approx 1.8 \times 10^{12} \text{ cm}^{-3}$ can be above the thresholds for both the shear and compressional modes.
- For the worst case of instantaneous slowing down, the current drive efficiency is reduced by an order of magnitude.
- A preliminary quasilinear calculation which includes several beamlines injected across the minor cross section each with a finite size footprint predicts a reduction in the fast ion current of order 25% for the present US/ITER proposed design.

NUMERICAL MODEL

- The code solves numerically the quasilinear equations:

$$\frac{\partial f_b(\mathbf{v}, t)}{\partial t} = C_p(f_b) + C_{ql}(f_b) + S(\mathbf{v}),$$

- The wave energy density obeys:

$$\frac{\partial W(\mathbf{v}, t)}{\partial t} = 2(\gamma_b + \gamma_e + \gamma_a)W(\mathbf{v}, t).$$

- The particle collision operator, $C_p(f_b)$, is linearized and includes slowing down, energy dispersion, pitch angle scattering, and is valid down to ion speeds $\approx 1.5 \times v_{th,i}$.
- The particle source $S(\mathbf{v})$ is proportional to:

$$S(\mathbf{v}) \propto \exp \left[- \left(\frac{v - v_b}{\Delta v_b} \right)^2 - \left(\frac{\mu - \mu_b}{\Delta \mu_b} \right)^2 \right]$$

- The quasilinear operator, $C_{ql}(f_b)$, is given by:

$$C_{ql}(f_b) = \frac{\partial}{\partial v_{\parallel}} \left(D_{ql} \frac{\partial}{\partial v_{\parallel}} f_b \right)$$

NUMERICAL MODEL (CONT)

- The diffusion coefficients for compressional and shear polarizations are given by:

$$D_{ql,comp} \propto \left(\frac{v_{\perp}}{v_{\parallel}} \right)^2 \frac{W(v_{\parallel})}{v_{\parallel}} J_1^2(k_{\perp}(k_{\parallel})\rho_b)$$

$$D_{ql,shear} \propto \left(\frac{v_{\perp}}{v_{\parallel}} \right)^2 \frac{W(v_{\parallel})}{v_{\parallel}} \left(\frac{v_D}{v_{\perp}} \right)^2 J_0^2(k_{\perp}(k_{\parallel})\rho_b)$$

- The shape of the spectrum, $W(v_{\parallel})$, is presently treated as a model. The models considered so far are:

1. Square functions centered where the instability drive is the strongest.
2. Functions very similar to those of Kulygin, Mikhailovskii, et. al. (Sov. Phys. JETP Vol. 43 1976) which spread out from the injection point as the distribution function flattens.

- The amplitude of W is determined from marginal stability, $\gamma_b + \gamma_e + \gamma_a \approx 0$
- The dispersion relation for each of the modes is used to relate $k_{\perp}$ to $k_{\parallel}$.

BENCHMARKS

- Moments of distribution function agree with analytic estimates in the appropriate limits. (Delta function source and $C_{ql} \rightarrow 0$)
- Fast ion current agrees with accepted results in the limit $C_{ql} \rightarrow 0$.
- Numerical diagnostics check particle and energy conservation for all values of C_p and C_{ql} .
- Growth rates of both modes for collisional distributions agree with those of Berk, Horton, Rosenbluth and Rutherford (Nuclear Fusion, Vol. 15, pp819-844 1975).

GENERAL RESULTS

- The beam footprint and the distribution of beamlines across the minor cross section are important considerations in determining the apparent source angular width and local source intensity, and hence the stability of the hot ion distribution function to Alfvén instabilities. Calculations assuming single pencil beams are pessimistic.
- The turbulence depletes the hot ion plateau in $v_{\parallel}$ from enhanced slowing down, and does *not* change the shape of the plateau significantly.
- The slowing down time due only to turbulence is usually comparable to the slowing down time due only to Coulomb collisions. Therefore the instantaneous slowing down model is pessimistic, and should not be used.
- If Alfvén instabilities are important in experiments injected with both super- and sub- Alfvénic particles (e. g. positive ion beams), the super-Alfvénic plateaus will be depopulated, and the sub-Alfvénic plateaus will remain largely unchanged. This may be a way to detect the presence of Alfvén instabilities experimentally.

COMPARISON WITH T-11 EXPERIMENTAL DATA

- Compressional branch of Alfvén wave is stable in T-11.
- The shear branch is unstable, and reduces the hot ion density by a factor of 1.5.
- Ratios of the levels of the sub-Alfvénic to super-Alfvénic plateaus from the T-11 data are indicative of an enhanced slowing down of the super-Alfvénic component.
(Data/Calculation reported by Leonov):

RESULTS FOR ITER

- The results of a preliminary calculation which includes the source angular width introduced from a finite footprint of the beam and several beamlines aimed at different flux surfaces, the fast ion current is reduced by about 25% for the US/ITER proposed design.
- Parametric studies are being performed to find a regime where Allvén instabilities are not present, at adequate beam densities to drive the required current.

ONGOING WORK

- Define low V_A experiments on D-III-D in collaboration with LLNL/GA group:
 1. Charge exchange diagnostics.
 2. Fluctuation measurements.
- Calculate the shape of the spectrum, $W(v_{\parallel})$, based on the local (in $v_{\parallel}$) instability drive.
- Develop a radial quasilinear code ($r - v_{\parallel}$) to study the global Alfvén mode.

- Finite beam footprint is a stabilizing influence on the Alfvén modes. The footprint introduces an angular width in the source.

US/ITER Parameters, $\theta_{inj} = 17^\circ$

Impact of Source Parameters on Compressional Mode Growth Rate

- The $v_{\perp} = 0$ distribution has a number of similarities with the T-11 data.
- The fast ion "current" is reduced by a factor of 1.7 by the shear Alfvén mode.
(Narrow angular width)

June 14, 1988

Hi Steve:

The package I'm sending along includes the following:

1) A figure showing how the growth rate computed from the collisional distribution varies as the energy and angular source widths are changed. This result is for the compressional mode, but the shear mode results are similar. This is for ITER parameters, at a mean injection angle of about 17 degrees. This should support the contention that widening the source helps, but only to a point. A factor of 10 reduction in growth rate can be easily achieved by the finite beam footprint.

2) Plots of f_b at $v_{\text{perp}}=0$ as a function of E_{parallel} for T-11. The source used for these is pencil-like, so I suspect these are slightly pessimistic. I've labeled the cases for the collisional distribution and the quasilinearly stabilized distribution. You can still see the depletion of the plateau, but because v_{perp} is approx $=0$, the half and third energy components are more pronounced (doesn't have as much weighting off v_{parallel} axis, where isotropizing influences are greater, $v_{\text{half}}=1.25 v_{\text{crit}}$, $v_{\text{third}}=v_{\text{crit}}$.) As you might have noticed, the collisional calculation looks very much like the Soviets calculation, suggesting that the $v_{\text{perp}}=0$ distribution is a pretty good indicator of the charge exchange signal. The depletion of the plateau from the turbulence is still there, but it is a matter of opinion on whether the data fits the quasilinear or the collisional results better. I still think the relative heights of the beam components is still suggestive. I was impressed with how my calculation looks so much like the charge exchange signal. I did no massaging of input data to get this, by the way.

3) Is the Soviet's global calculation a stability calculation using the collisional distribution, or a quasilinear calculation which evolves the distribution function? I suspect that it is only a stability calculation, analogous to the problem that Fu and VanDam at IFS did for the alpha particles. Also since the guy who did it is an ICRH proponent, he would have the tools readily available to do the radial field structure. Could you emphasize that doing the stability calculation is an important first step, whether one looks at local or global modes, but a quasilinear calculation must be done in either case to assess the impact on

the distribution, and therefore the current drive efficiency!!!
The growth rates I get from the collisional distribution are very high also (5×10^5 1/s), and introducing the turbulence drives the growth rate down to a point where the slowing down is comparable to that due to classical collisions. Relying on only the initial growth rate can be misleading. Mention (if you didn't get to that viewgraph) that I am beginning to work on a code to do the global problem (did you know that the Cray III is coming to NMFECC later this year?) .

- The $v_{\perp} = 0$ distribution has a number of similarities with the T-11 data.
- The fast ion "current" is reduced by a factor of 1.7 by the shear Alfvén mode.
(Narrow angular width)

• Collisional Distribution $f(v_{\perp} = 0, v_{\parallel})$

• Quasilinearly Stable Distribution $f(v_{\perp} = 0, v_{\parallel})$

- Finite beam footprint is a stabilizing influence on the Alfvén modes. The footprint introduces an angular width in the source.

US/ITER Parameters, $\theta_{inj} = 17^\circ$

Impact of Source Parameters on Compressional Mode Growth Rate

June 30, 1988

MEMORANDUM

To: ITER Physics Distribution

From: R.B. Campbell

Subject: Comments on Soviet Alfvén Wave Stability Calculation

1 Summary

The stability calculation presented by K.N. Stepanov at the Current Drive and Heating Workshop indicated that a strong instability could exist in ITER which might preclude the use of neutral beams for current drive. This conclusion was arrived at by a global linear stability analysis which predicted very strong growth of a mode with a frequency just below ω_{ci} .

The claim I am making in this note is that the model distribution function he used, which has zero width in pitch angle, results in pessimistic answers. I have evaluated some of the pertinent terms in the stability theory for a realistic ITER beam driven distribution and conclude that the growth rates will be lower than what Stepanov indicates.

Regardless of the size of the linear growth rate, a quasilinear calculation is necessary to determine the final effect on the beam heated distribution. I have the capability, after relatively small code modifications, to look at this mode and assess its effect on current drive.

2 Introduction

K.N. Stepanov gave a presentation "NBI Driven Alfvén Instability" at the Current Drive and Heating Workshop held in Garching June 13-17, 1988. I have received a copy of the viewgraphs from Steve Devoto, and all my comments are based on reading these notes.

3 Analysis

First of all, the mode considered by Stepanov ($\omega \leq \omega_{ci}$) is quite different than the low frequency ($\omega \ll \omega_{ci}$) shear and compressional modes I have been looking at. His mode is the ion analogue to the Pogutse-Pareil (PP) instability, whereas the modes I have studied resemble long wavelength MHD modes. Berk, et. al.¹ determined that the lower frequency (smaller $k_{||}$) modes were more important to the viability of the two component tokamak (TCT) being proposed at the time. This makes sense, because lower frequency long wavelength "global" modes are tougher to stabilize and have more opportunity to effect the gross properties of the plasma.

The PP instability was discussed by Berk, et. al. in the context of beam free energy coupling to the Whistler branch ($\omega \geq \omega_{ci}$). They concluded that the mode would not be important in TCT because to go unstable, it required that the beam energy exceed the Alfvén energy by a factor of 9. For a 5 meter ITER, this factor becomes about 7.2 suggesting that the beam energy would have to exceed 2.2 MeV before the mode would manifest itself. For the 5 meter ITER proposal, the beam energy is 1.6 MeV, somewhat below this energy. If the US/ITER were to go to 5.8 meters, a preliminary estimate suggests that the required beam energy must be 2.1 MeV, considerably closer to this minimum energy, and perhaps a cause for concern.

Stepanov's calculation deals with the slow branch, ($\omega \leq \omega_{ci}$). I do not know why Berk did not look at coupling to this frequency mode, because it seems to have a minimum resonant parallel energy much closer to the Alfvén energy than the Whistler.

For either mode, there is a criterion for instability relating parallel and perpendicular velocity space gradients. This was derived by Berk in his paper, and is given by:

$$\left\langle \frac{v_{\perp}^2}{v_{\parallel}^2} \right\rangle \leq \frac{\omega_{ci}}{\omega_{ci} + \omega}, \quad (1)$$

where ω is the mode frequency, and ω_{ci} is the beam cyclotron frequency. This relationship is essentially the Sagdeev-Shafranov criterion discussed by Stepanov in his presentation, integrated over $v_{\perp}$. The term in the angle

¹Berk, et. al. "Microstability Theory of Two-Energy-Component Toroidal Systems" *Nuclear Fusion*, 15, p819 (1975)

brackets is a $v_{\perp}$ moment of the distribution function, f_b , evaluated at the resonant $v_{\parallel, res} = \frac{\omega + u_{\perp}}{k_{\parallel}}$, and is given by:

$$\left\langle \frac{v_{\perp}^2}{v_{\parallel}^2} \right\rangle = - \frac{1}{\int_0^{\infty} du_{\perp} u_{\perp} f_b(u_{\perp}, u_{\parallel, res})} \int_0^{\infty} du_{\perp} u_{\perp}^3 \frac{\partial f_b}{\partial u_{\parallel, res}}. \quad (2)$$

As far as I can tell, Stepanov did not evaluate Eq. 2 for an ITER case with a source with a finite angular width. (He used a Delta function in $\cos \theta$.)

I have evaluated this moment for the ITER collisional distribution using a reasonable amount of angular spread caused by the finite beam footprint for ITER and the design appears to be at the boundary for instability defined by Eq. 1 for the $\omega \leq \omega_{ci}$ branch ($\omega \approx 0.6\omega_{ci}$). This result warrants our examination of this mode more carefully in the context of the quasilinear calculation. Unless the beam energy approaches 2.2 MeV, the whistler branch does not seem to be important.

I agree with Stepanov's statement about having to do a radial analysis of this mode, because of its high growth rate. He only looked at the linear stability of the mode. A quasilinear calculation needs to be done. I will be developing a simple code next year which will perform the quasilinear calculation in a $(r - v_{\parallel})$ coordinate space which will address some of the self-consistent aspects of these PP-type modes, as well as the low frequency global Alfvén modes.

I now would like to demonstrate how Stepanov's assumption of a delta function in angle results in a pessimistic estimate for equation 2. I have evaluated equation 2 using the same distribution he uses:

$$f_b(v, \cos \theta) = \frac{C}{v^3 + v_c^3} \times \delta(\cos \theta - \cos \theta_0) \quad (3)$$

The result for $\left\langle \frac{v_{\perp}^2}{v_{\parallel}^2} \right\rangle$ is:

$$\left\langle \frac{v_{\perp}^2}{v_{\parallel}^2} \right\rangle = \frac{1}{2} \sin^2 \theta_0 \left[\frac{3}{1 + \left(\frac{v_c \cos \theta_0}{v_{\parallel, res}} \right)^3} - 1 \right] \quad (4)$$

In these equations, θ_0 is the injection angle, $v_{\parallel, res}$ is the resonant phase velocity, and v_c is the critical speed. Notice that this moment vanishes

exactly for tangential injection, making the criterion for instability well satisfied. Even for an injection angle of 20° , the mode would be unstable for such a narrow distribution.

4 Conclusions and Recommendations

Stepanov's distribution assumption results in growth rates which are clearly pessimistic. A larger value for eq. 2 which correspond to a distribution with less anisotropy will result in lower growth rates. Therefore, the growth rate calculation should be performed for this mode using a more realistic distribution function. At that point, if the growth rate exceeds the electron damping, a quasilinear calculation is warranted in which an estimate of the degradation in current drive efficiency can be made. If my previous results on the low frequency shear and compressional modes are any indication, a large linear growth rate does not necessarily mean a large reduction in current drive. For ITER, with linear growth rates of $\gamma \approx 5 \times 10^5 \text{ sec}^{-1}$ for the shear mode, the final reduction in current drive efficiency is only about 25 %.

Abstract Submitted for the Thirtieth Annual Meeting
Division of Plasma Physics
October 31 - November 4, 1988

Category Number and Subject 3.1 C D with NBs

☒ Theory ☐ Experiment

Alfvén Wave Stability in Neutral Beam Driven Tokamaks.* R. B. CAMPBELL (TRW/LLNL), and R. S. DEVOTO (LLNL) — Neutral beam current drive in the US/ITER proposal calls for particles injected at energies between 1 and 3 MeV, above the central Alfvén energy of 300keV. Velocity space driven instabilities can be present, which could reduce current drive efficiencies due to enhanced slowing down of injected particles. The most important waves are the low frequency ($\omega \ll \omega_{ch}$) shear and compressional modes, as well as a higher frequency ($\omega \leq \omega_{ch}$) anomalous Doppler mode. We have developed a quasilinear code which determines the ion distribution function in the presence of these instabilities. We have used the code to interpret experimental data from the Soviet T-11 tokamak during injection of tangential super-Alfvénic beams. We have noticed similarities between the measured and calculated high energy plateau, when the calculation includes the quasilinear effects. We have also applied the model to the US/ITER, and preliminary results indicate that the fast ion current driven in a plasma influenced by the low frequency Alfvén waves is about 25% lower than the calculation including only Coulombic slowing down. Additional results will be reported at the conference.

* Work performed for USDoE by LLNL under Contr. W-7405-ENG-48, and by TRW under Subcontract 2375703 with LLNL

- ☒ Prefer Poster Session
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Submitted by:

(Signature of APS Member)

Robert B. Campbell
(Same Name Typewritten)

LLNL, L-644
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94550

(Address)

This form, or a reasonable facsimile, plus **TWO XEROX COPIES**, must be received by **NO LATER THAN NOON**, Friday, July 15, 1988 at the following address:

Ms. Barbara Sarfaty
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Commercial: (609) 243-2440
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James Forrestal Campus
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August 27, 1988

MEMORANDUM

To: ITER Physics Distribution

From: R. B. Campbell

Subject: Why Does Alfvén Wave Turbulence Reduce the Current Drive Efficiency?

SUMMARY

A question has been raised as to why there is *any* degradation of the current drive efficiency due to the excitation of Alfvén waves. The simple picture of why the Alfvén wave turbulence degrades the current drive efficiency relies on the notion that the interaction of the waves with the particles results in enhanced slowing down of the hot injected particles. Because of the diffusive nature of the quasilinear operator which describes the wave-particle interaction it is conceivable that the turbulence might instead flatten the distribution function and thereby produce no additional slowing down over the irreducible Coulomb contribution.

I will present a general derivation in this note which shows that while the flattening of the distribution will occur, it cannot occur without some slowing down of the hot beam injected particles. The essence of the argument is that for a given source of hot injected ions, the slowing down flux must be the same regardless of whether it is due to Coulomb or anomalous processes. The Coulomb slowing down flux is primarily advective, meaning it is proportional to the distribution function. When an additional flux is introduced due to the Alfvén instability, (which can manifest itself even at low turbulence levels) the Coulomb flux must decrease. The only way for this to happen, recalling the advective nature of the operator, is for the local distribution function to decrease.

The result derived here also applies to electron cyclotron and lower hybrid heating and current drive, although in those cases the tendency is to *enhance* the hot tail density. This is because the distribution (Maxwellian)

which is being acted on by the waves initially has a negative velocity space derivative, the flow of particles is then to higher energy.

The results from the SFP code which I have developed support the idea of enhanced slowing down of the injected particles due to the Alfvén waves. For the US/ITER with a 5 meter major radius, the reduction in current drive efficiency predicted by the SFP code is a tolerable 25% .

INTRODUCTION

Why should there be any degradation of the current drive efficiency due to the excitation of Alfvén waves? The reasoning behind this question is that the first tendency of the hot particle distribution function under the influence of the turbulence may be to flatten, thereby removing any anomalous slowing down due to the waves. This result follows from the fact that the magnitude of the quasilinear operator depends only on the first and second derivatives of the distribution, not the distribution function itself. In contrast, the particle flux due to Coulomb collisions in this region of velocity space has a term proportional to the distribution function, the so-called "advective" term, and hence will always have a non-zero contribution to the slowing down, regardless of how flat the distribution function gets.

A simple analysis of the quasilinear equation presented here (see eqns. 4 to 6) indicates that the hot tail density must drop while the distribution flattens.

ANALYSIS

It would be desirable to have a simple model to illustrate why some slowing down must exist during the flattening process. Without loss of generality, the steady-state quasilinear equation can be written:

$$C_{coll}(f) + C_{qt}(f) = -S(v) \quad (1)$$

The Coulomb collision operator takes the form:

$$C_{coll}(f) = \nabla_v \cdot (D_{coll} \cdot \nabla_v f + \tilde{A} f), \quad (2)$$

and the quasilinear operator is given by:

$$C_{qt}(f) = \nabla_v \cdot (D_{qt} \cdot \nabla_v f). \quad (3)$$

The functions D_{coll} and D_{qt} are the diffusion tensors associated with the fluxes due to Coulomb collisions and wave-particle interactions, respectively. The vector $\tilde{A}$ is the function of proportionality between the distribution function and the advective part of the collisional flux. Because of the purely-diffusive nature of the quasilinear operator, eq. 3, it is conceivable that the distribution function could flatten without the reduction of the level of the hot distribution function. A simple argument, based on the above equations, will show that some reduction of the magnitude of the distribution function is necessary.

I have considered two cases, one in which there is no effect from the Alfvén wave turbulence (“classical”), and another in which the distribution is affected by quasilinear diffusion (“anomalous”). I denote the distribution function in these two cases as f_{class} and f_{anom} respectively. I have the option of comparing the two cases at the same hot density or the same hot particle source strength. For computing the change in the fast ion current in the presence of the Alfvén instability and the corresponding change in $\frac{I}{\rho}$, the natural choice is to keep the beam source strength (e. g. beam power) constant, and compute the change in the density. With this choice, I write the appropriate forms of eq. 1 governing f_{class} and f_{anom} , subtract them, and apply the divergence theorem over a surface enclosing the source. The result is:

$$\int_{S(v)} d\tilde{S} \cdot n \cdot [D_{coll} \cdot \nabla_v f_{class} + \tilde{A} \cdot (f_{class} - f_{anom}) - (D_{coll} + D_{qt}) \cdot \nabla_v f_{anom}] = 0. \quad (4)$$

This equation has a simple physical meaning. It says that the steady-state flux of particles through any surface enclosing either all or part of the particle source must be the same for both cases, regardless of whether the slowing down and dispersion is due to classical or anomalous processes. For the neutral beam problem I am considering, the particle collision operator is primarily advective in the $|\vec{v}|$ variable. For this case, the flux relationship becomes more transparent:

$$\int_{S(v)} d\tilde{S} \cdot n \cdot [\tilde{A} \cdot (f_{class} - f_{anom}) - D_{qt} \cdot \nabla_v f_{anom}] = 0. \quad (5)$$

By changing the position and shape of the surface S , we can see how the difference between f_{class} and f_{anom} changes in in all the velocity space variables.

If I expand f_{anom} in a small parameter ϵ which represents the strength of the rf interaction compared to the collisional interaction, I can demonstrate how the distribution function changes locally as the turbulence level is increased. The distribution is expanded as $f_{anom} \approx f_{class} + \epsilon \delta f_\epsilon + \dots$, and the diffusion tensor is taken as $D_{qi} \approx \epsilon D_{1,qi}$. Equation 5 then becomes to first order in ϵ :

$$\int_{S(\mathbf{v})} d\tilde{S} \, \mathbf{n} \cdot \left(\tilde{\mathbf{A}} \cdot \delta f_\epsilon + \mathbf{D}_{1,qi} \cdot \nabla_{\mathbf{v}} f_{class} \right) = 0. \quad (6)$$

For simplicity, I take the surface to be one of constant $|\tilde{\mathbf{v}}|$, where the relevant component of $\tilde{\mathbf{A}}$ is $\nu_{coll} \frac{(1+v^2)}{u^2}$, and the quasilinear flux for nearly tangential injection is $D_{1,qi} \frac{\partial f_{class}}{\partial v}$. For the beam injection problem, the derivative for the collisional distribution, f_{class} is positive in the region of instability (where D_{qi} is non-zero)¹. Therefore, the first order correction δf_ϵ is negative, indicating that the hot density must *decrease* even for low levels of turbulence.

The distribution will flatten as the strength of the rf is increased and the rf-driven flux saturate, but it takes a complete quasilinear treatment such as implemented in SFP to determine the ultimate reduction in beam density due to the instability.

It is important to note that the above arguments can also be applied to the case of heating and current drive by externally applied waves. In this case, the distribution analogous to f_{class} is Maxwellian, and has a negative derivative. As the rf is turned on, the energetic density *increases* in the range where the waves are applied, corresponding to the creation of a high energy tail. Substantial hot density increases to drive current can be sustained in this way. The hot density increase is limited by the tendency of the distribution function to flatten and the rf flux to saturate in the limit of large diffusion coefficient. To say that the hot density does not decrease when Alfvén wave turbulence is present in the beam injection problem is equivalent to saying that an energetic tail is not produced when lower hybrid waves are applied for current drive purposes. Extensive theoretical and experimental work on lower hybrid current drive bears out the existence of the enhanced tail density in this case.

¹ Because D is non-zero in a band of $v_{||}$, the surface integral of $|\tilde{\mathbf{v}}| = \text{const}$ involves an integral over a limited range of μ . In this range of μ , the average derivative is positive.

Note that particle conservation is being obeyed. For the beam problem, the density "lost" by the hot distribution shows up in a slight increase in the bulk ion distribution. For the ECH and LH problem, the increased density is provided by a source of bulk particles heated by the waves.

To illustrate the effects I have described above, I have exercised SFP on two model beam injection problems. For the first one, I have introduced turbulence with phase velocities below the injection velocity, and above the Alfvén speed. This is practically the same as the actual case for the stability problem. The second model case considers turbulence introduced from slightly below the injection speed and extends *above* the injection speed. This case is analogous to the lower hybrid current drive problem (except with ions), although the source of particles is from the injected beam and not from the bulk distribution.

Consider first the case of the turbulence below the injection speed. The only difference from the actual case I consider for the Alfvén problem is that in the model problem no turbulence exists within the velocity range where source particles are introduced. The results are shown on Figure 1. In this figure, I plot both f_{class} (solid) and f_{anom} (dotted) as a function of v for a single value of $\bar{\mu}$ equal to the injection $\mu = \mu_{inj}$. The region where the rf diffusion is non-zero is also shown. The examination of a single pitch angle from the full two-dimensional solution which I calculate is really only part of the picture, because the diffusion is two-dimensional. The two-dimensionality of the problem comes in because flux across a boundary which was directed perpendicular to a constant $|\bar{v}|$ surface in the collisional problem will now have a significant component along a line of $v_{\perp} = const$ due to the rf diffusion along characteristics.

The strength of the diffusion is typical of that required to make the growth rate of the shear mode equal to the damping rate due to the electrons. This level is typically below that which is required for substantial flattening of the distribution. The reduction of f in the region where turbulence is being introduced is due directly to the effects described above in equation 6. The reduction in the distribution above and below the region of rf diffusion decrease to insure continuity of the distribution function. Note that the shape of the distribution stays the same in the rf diffusion-free region because the quasilinear equation in that region does not differ from the collisional kinetic equation.

Figure 2 shows the results of the second model problem where the turbu-

Figure 1: Comparison of f_{close} and f_{anom} at the injection pitch angle, μ_{inj} . The region where tf diffusion is introduced is also shown.

Figure 2: Comparison of f_{class} and f_{anom} at the injection pitch angle, μ_{inj} . The region where rf diffusion is at and above the parallel injection speed.

lence is introduced from the parallel beam injection speed to higher phase velocities. In reality, this turbulence could not be maintained from the free energy of the distribution, since it assumed to have phase velocities away from the destabilizing part of the distribution. The arrangement of the source particles and the waves has similarities to the lower hybrid current drive case. The energetic density increases over the purely collisional case, particularly above the injection velocity. The waves in this case heat the particles, scattering them from lower parallel velocities to higher. The mechanism can be seen from the simple model eq. 6 since the appropriate surface averages are of the opposite sign when compared to the Alfvén problem.

C TETRA Systems Code Development and Application

APPLICATIONS OF DECISION-ANALYSIS TECHNIQUES
TO THE SYSTEMS CODE

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TETRA Systems Code Meeting

FEDC

March 9, 1988

(dj803078/ljpv)

OBJECTIVES

- (1) To determine better Value Functions (i.e., Figures-of-merit) for ITER System Studies than simply Capital Cost.
- (2) To develop a methodology for ITER which will aid the consensus process in making design selection decisions.

WHAT IS WRONG WITH CAPITAL COST ALONE?

Nothing -- if it's your only consideration.

However, in ITER, the problems with capital cost alone are:

- (1) Critics of optimization procedures with a single figure-of-merit (i.e., cost) say we don't need an optimization code; simply set all the important constraints at their bounds and solve a deterministic set of equations (i.e., where number of variables = number of equations).
- (2) "Reasonable Cost" is only one of many requirements of the ITER Terms-of-Reference; we need to look at all important attributes.
- (3) Using cost alone does not address the trade-offs with other important requirements.

WHICH IS THE BEST ITER DESIGN?

	<u>I</u>	<u>II</u>	<u>III</u>
Capital Cost (B\$)	3	4	6
Prob. of non-ignition	33%	10%	4%
Av. wall loading (MW m ⁻²)	2.0	1.3	1.0
Peak Fluence (MW yr m ⁻²)	2.0	1.0	0.3
Q (current-drive)	5	10	20

METHOD

1. Determine a set of uncorrelated attributes which provide the best measure of the goals and objectives of the design.
2. Assign a relative value scale to each attribute..
3. Combine all attributes into a single composite value function (i.e., Figure-of-merit).
4. Characterize the uncertainty on major input variables. (Our methodology will ensure that we will maximize the expected "value" over these uncertainty distributions).

ITER OBJECTIVES
FROM THE ITER TECHNICAL-WORKING-GROUP TERMS OF REFERENCE,
SEPT-DEC 1987

Basic Objective:

"A single conceptual design of an ITER which would, for reasonable cost, demonstrate the scientific and technological feasibility of fusion and provide the database in physics and engineering necessary for the design and construction of a demonstration fusion plant."

Sub-Objectives

1. Demonstrate, and complete database for, controlled ignition and extended burn of DT plasma.
2. Confirm database for ultimate steady-state operation.
3. Demonstrate technologies essential to a prospective reactor in an integrated system, including high heat flux components, nucl. components and s/c magnets.
4. Demonstrate maintainability and assess component + plant reliability/ availability.
5. Demonstrate safe and environmentally acceptable operation.
6. Test tritium and energy extraction in local, reactor relevant modules.
7. Perform material testing appropriate to the DEMO.
8. Provide flexibility for the introduction of advanced operating modes and features which increase the reactor attractiveness of ITER.

ITER REQUIREMENTS AND CONSTRAINTS

1. PHASES:

- (i) Physics: Achievement of physics objectives (pulsed ignition and, if possible, steady-state).
- (ii) Technology: Achievement of engineering objectives and testing program.

2. IGNITION:

ITER will initially be operated in pulsed (ignited) mode where external power (for profile control, etc.) is insignificant^{compared} to the alpha power.

3. STEADY STATE:

During physics phase, ..., extend burn towards steady-state with as high a Q as possible, ..., through NB and/or RF techniques. During ~~physics~~^{Technology} phase..., should aim for steady-state, even if it only allows $Q \sim 5$.

4. CONFINEMENT:

Sufficient to reach ignition ... which necessitates $I \sim 10\text{-}20\text{ MA}$, ... with appropriate q_{edge} and elongations < 2 .

5. BURN PULSE LENGTH:

Sufficient to reach equilib. with thermalized ash/impurities, ..., τ_{burn} should be as large a fraction of the global skin time as possible (— equilibrium profiles, $\tau_{\text{burn}} \sim \text{few hundred secs}$).

ITER REQUIREMENTS AND CONSTRAINTS (continued)

6. INDUCTIVE CAPABILITY:

Capability for pure inductive rampup and for minimum τ_{burn} .
(RF techniques can be employed to aid rampup and extend burn but not at expense of inductive capability).

7. NEUTRON WALL LOAD:

$> 1 \text{ MW m}^{-2}$ (av.)

8. END-OF-LIFE FLUENCE:

Useful fluence $\sim 1 \text{ MW yr m}^{-2}$. Machine should be designed for $\sim 3 \text{ MW yr m}^{-2}$.

9. TRITIUM BREEDING:

Technology phase should aim for TBR as close to unity as possible (can be of a non-reactor-relevant design).

10. MACHINE AVAILABILITY:

To achieve fluence of 1 MW yr m^{-2} in reasonable operating lifetime ($\sim 10 \text{ yr}$), availability in technology phase must be $\sim 10\%$. Also ultimate demonstration of steady-state should include a continuous ~~time~~ operation of $\sim 1 \text{ week}$ or more.

CANDIDATE ATTRIBUTE SET FOR ITER SYSTEM STUDIES

Initial Set

1. Total Capital Cost (M\$)
2. Probability of non-ignition (net ignition margin)
3. Average neutron wall loading (MW m^{-2})
4. Peak neutron fluence^a (MW yr m^{-2})
5. Q for current-drive operation

Near Term Extensions

6. Safety and environment^b
7. Availability and reliability
8. Access and test area/volume
9. Flexibility (operational and flexibility to meet mission goals^c)

^a Requires some form of availability model (see #7 above) or, for pulsed operation, a burn pulse length and cycle limit

^b See recent work by Piet and Brereton

^c Might be better to restate in terms of "Risk".

SCALAR ATTRIBUTE UTILITY FUNCTION

$$u(\text{flux}) = 1.01 * (1 - \exp(-1.72 * (\text{flux} - 0.5)))$$

SCALAR ATTRIBUTE UTILITY FUNCTION

$$u(\text{cost}) = 0.396 * (7.0 - \text{cost}) \sim 0.576$$

SCALAR ATTRIBUTE UTILITY FUNCTION

$$u(Q)=1.00*(1-\exp(-0.344*(Q-4.5)))$$

SCALAR ATTRIBUTE UTILITY FUNCTION

$$u(\text{flue}) = 1.00 * (1 - \exp(-1.15 * (\text{flue} - 0.3)))$$

SCALAR ATTRIBUTE UTILITY FUNCTION

$$u(I) = (0.3 - (\text{prob. of no ignition}) / 0.3$$

INDIVIDUAL VALUE FUNCTIONS

<u>Attribute</u>	<u>Range</u>	<u>Function</u>
1. Cost	2 - 7 B\$	$U_{\$} = 0.396 (7.0 - \$)^{0.576}$
2. Prob. of non-ignition	$0^{+} - 0.3$	$U_I = (0.3 - P(\bar{I}))/0.3$
3. Wall load	$0.5 - 3.5 \text{ MW m}^{-2}$	$U_{\Gamma} = 1.01 (1 - \exp (-1.72(\Gamma-0.5)))$
4. Fluence	$0.3 - 6 \text{ MW yr m}^{-2}$	$U_f = 1 - \exp (-1.15(f-0.3))$
5. Q current-drive	4.5 - 20.0	$U_Q = 1 - \exp (-0.344(Q-4.5))$

COMPOSITE VALUE FUNCTION

$$U = \underbrace{k_{\$ \$} U_{\$ \$}}_{\text{Cost}} + \underbrace{k_{PI} U_{PI}}_{\text{Performance/Ignition-risk}}$$

$$U_{PI} = \underbrace{k_P U_P}_{\text{Performance}} + \underbrace{k_I U_I}_{\text{Ignition}} + (1 - k_P - k_I) U_P U_I$$

$$U_P = \underbrace{k_{\Gamma \Gamma} U_{\Gamma \Gamma}}_{\text{Wall-load}} + \underbrace{k_F U_F}_{\text{Fluence}} + \underbrace{k_Q U_Q}_Q$$

$$\begin{aligned} [k_{\$ \$} &= 0.175, & k_{PI} &= 0.825; \\ k_F &= 0.57, & k_I &= 0.19; \\ k_{\Gamma} &= 0.48, & k_f &= 0.28, & k_Q &= 0.24] \end{aligned}$$

JUNE 1, 1978

SYSTEMS CODE DESIGN STUDIES

D. T. BLACKFIELD / L. J. PERKINS

- MINIMUM COST FOR VARIOUS SCALING LAWS
- MINIMUM COST FOR FIXED PLASMA CURRENT,
FOR VARIOUS PHYSICS / ENGINEERING
ASSUMPTIONS

ENGINEERING ASSUMPTIONS

SCRAP LI = 20 cm

SCRAP LO = 10 cm

INBOARD SHIELD = 70 cm

INBOARD FIRST WALL = 3.5 cm

OUTBOARD SHIELD = 70 cm

OUTBOARD FIRST WALL = 3.5 cm

PHYSICS ASSUMPTIONS

NO DENSITY LIMIT

RESISTIVE STARTUP VOLT-SECONDS = 30 VS

(NO 1.6 LI REQUIREMENT)

$$Z_{eff} = 1.78$$

$$\text{KAYE-BIG } \tau_e = f \times 0.124 \times (A_{nuk})^{1/4} I_p^{.25} n_e^{-1} B_T^{.3.3} a R^{1/2} / P_{HEAT}^{1/2}$$

MINIMUM COST DESIGNS

VSRESIN = 30; TBURN = 300 s; Zeff= 1.777; Zimp 10.66;

Aimp = 20.; RALPNE = .04465; CSAWTH = 1.

CASE	5.2	5.7	6.1	6.3	6.5	6.7
R (m)	5.45	4.88	5.14	5.84	5.29	6.88
a (m)	1.27	.939	2.06	1.01	1.93	2.75
B (T)	6.83	7.00	4.3	7.7	5.19	4.78 5.33
I (MA)	12.0	7.5	21.0	7.9	21.6	31.2 34.7
BMAXTF (T)	12	11.9	11.0	12	12	12
BMAXOH (T)	3.6	3.1	8.0	1.88	8.09	5.96
Pfus (MW)	926	445	1555	565	1153	2385 2975
Gamma (MW/m2)	1.43	1	1.6	1	1.25	1.4 1.8
Te (keV)	7.50	9.2	9.3	10.3	9.1	14.2 14.4
SCALING	K-G	GOLDS	JAERI	T-10	BIG	T-10 JAERI
H-FACTOR	1.3	2.0	1.5	1.5	2.1	1.5 1.5
DIRECT (M\$)	1868	1477	2146	1753	2138	4202
TF (M\$)	361	264	244	390	353	727
PF (M\$)	217	180	296	208	283	1017
FDENE	.932	.65	.94	.62	.74	.64 1
FBETA	.786	.924	1	.84	.68	.80 1
FVS	1	1	1	1	1	.91
FTFSTS	.75	.75	.75	.75	.75	.75
FOHSTS	.10	.10	.33	.71	.26	.20
FEPTF	1	1	1	1	1	1
FEPOH	.41	.54	.87	.16	.60	.39
FCPTTF	1	1	1	1	1	1
FCPTOH	1	.94	.88	1	1	1
FCPTTFQ	.50	.56	.92	.44	.65	.54
FBMAXPF	.65	.69	.62	.65	.58	.74
FTFACND	1	.93	1	1	1	1

20 MA PLASMA CURRENT
GOLDSTON SCALING (H = 2.0)

CASE	5.11	7.1	7.2	7.3	7.4	7.5	7.6	7.7
R (m)	4.98	5.74	5.49	5.34	5.10	5.03	4.95	4.89
a (m)	1.21	1.93	1.90	1.88	2.03	1.99	1.96	1.95
B (T)	4.45	5.22	5.15	5.11	4.20	4.29	4.36	4.35
BMAXTF (T)	11.3	11.0	11.2	11.4	10.7	10.7	10.7	10.8
BMAXOH (T)	9.3	8.5	9.5	9.7	9.45	9.16	8.97	9.36
SCRAPLI (cm)	20	20	20	20	20	15	10	10
Troyon Coeff.	2.8	1.8	1.8	1.8	3.0	2.8	2.8	2.8
KAPPA	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.4
TRIANG	.6	.6	.6	.6	.6	.6	.6	.67
FTFSTS	.75	.4	.45	.5	.5	.5	.5	.5
FOHSTS	.23	.4	.42	.43	.44	.46	.48	.48
Pfus (MW)	871	996	938	905	926	889	850	898
Gamma (MW/m2)	1	1	1	1	1	1	1	1
Te (keV)	11.1	5.6	5.8	5.9	10.1	10.0	9.7	10.9
DIRECT (M\$)	1861	2452	2279	2164	1994	1952	1912	2030
YF (M\$)	243	631	526	454	301	296	290	304
PF (M\$)	257	296	289	281	279	267	256	332
FDENE	.604	1.24	1.15	1.10	.69	.68	.68	.61
FBETA	.817	1	1	1	1	1	1	1
FVS	1	1	1	1	1	1	1	1
FEPTF	1	1	1	1	1	1	1	1
FEPOH	.90	.91	.93	.94	.97	1	1	1
FCPTTF	1	1	1	1	1	1	1	1
FCPTOH	.90	1	1	1	1	1	1	1
FCPTTFQ	.86	.42	.46	.50	.61	.61	.61	.63
FBMAXPF	.63	.53	.55	.57	.62	.63	.65	.68
FTFACND	1	1	1	1	1	1	1	1

NEW 5.5 ITER BASELINE -- CURRENT DRIVE
PERFORMANCE AND SENSITIVITY TO COMPARISON

- NB+LH+ bootstrap current-drive • $I_{ES}/I = 0.3$
- $R = 5.5m$, $a = 1.8m$, $B = 5.30T$ • wall load = $1MW/m^2$

OLD EBD $R=5.5m$ $I=20MA$	20
	3.0
	24/13
	0.663
	58.1
	20.9
	18.9
	13.2
	(14.94)
	1.84
	0.90
	0.77
	1.04

	REFERENCE BASELINE ($H_0 = 1.84$)	$H_0 = 1.7$	$H_0 = 1.6$	$H_0 = 1.5$
$I (MA)$	18	18	18	18
I_{Troyon}	3.0	2.89	2.81	2.69
$I_C / T_i (keV)$	23.2/21.8	21.5/19.6	20.1/18.0	18.6/16.4
$n_e (10^{20} m^{-3})$	0.654	0.690	0.724	0.765
$P_{NB} (MW)$	54.3	58.5	62.8	70.1
$P_{LH} (MW)$	20.9	24.5	29.1	33.6
$P_{total} (MW)$	75.1	83.5	91.9	103.8
Q	11.9 (12.7)	10.7	9.75	8.64
H_0	1.84	1.70 (fixed)	1.60 (fixed)	1.50 (fixed)
H_3	2.04	1.89	1.77	1.63
H_{T-10}	1.74	1.63	1.54	1.46
H_{R-L}	1.04	0.94	0.86	0.79

EXPERIMENTAL RESULTS
FOR ITER...

"ITER" Parameters $R=5.5m$ $I=20MA$ $I=22MA$	"ITER" Parameters $R=5.5m$ $I=22MA$
1.73	1.72
1.98	1.69
1.79	1.70
1.03	0.80

* Q value for baseline using $I_{ES} = f \sqrt{B} / R = 0.25$

BDE PARAMETERS GENERATED BY TETRA BI-MODAL

D.T. BLACKFIELD LLNL/TRW

L.J. PERKINS LLNL

JUNE 22, 1988

BDE PARAMETERS GENERATED BY TETRA

GOLDSTON SCALING (H = 1.5)

CASE	15.4	15.5	CASE	15.4	15.5	
R (m)	5.80	5.80	FDENE	.530	.924	fraction of density limit
a (m)	2.00	2.00	FBETA	.812	1	fraction of beta limit
A	2.9	2.9	FVS	1	1	V-S required / V-S supplied
B (T)	5.10	5.10	FTFSTS	.75	.75	fraction of TF stress limit
I _p (MA)	20.0	20.0	FOHSTS	.269	.272	fraction of OH stress limit
ELONGATION	2.2	2.2				
TRIANG.	0.6	0.6				
			FEPTF	1	1	frac. of TF stability margin
Pfus (MW)	1042	2085	FEPOH	.140	.141	frac. of OH stability margin
Gamma (MW/m2)	1	2	FCPTTF	1	1	frac. of max. TF cur. density
Pdiv (MW)	125	250	FCPTOH	1	1	frac. of max. OH cur. density
I _e (keV)	14.0	10.6	FCPTTFQ	.605	.612	frac. of max. TF quench press
n _e (x10 ²⁰ /m3)	.716	1.25	FBMAXPF	.510	.528	frac. of max. field on PF coil
nDT (x10 ²⁰ /m3)	.602	1.05	FTFACND	.603	.601	frac. of max. JBR on TF cond.
Z _{eff}	1.78	1.78				
beta (%)	4.46	5.49	BMAXTF (T)	11.1	11.1	
Troyon coef.	2.8	2.8	NO. OF TF COILS	16	16	
safety factor	.879	.879				
			BMAXOH (T)	6.17	6.17	
DIRECT (M\$)	2332	2663	BMAXPF (T)	5.61	5.81	
TF (M\$)	482	482	V-S CAPABILITY	189	192	
PF (M\$)	290	309	EF V-S	111	114	
VACUUM SYS (M\$)	129	348	OH V-S	78	78	
HEAT TRANS (M\$)	91	157	V-S BURN	9.44	9.61	
			BURN TIME (s)	300	300	

ENHANCMENT FACTORS REQUIRED FOR IGNITION MARGIN = 1

KAYE-GOLDSTON	1.66	1.37
GOLDSTON	1.77	1.60
KAYE-BIG	2.59	2.21
JAERI	1.94	1.43
T-10	1.76	1.57
PF1001	0.96	0.65

BDE PARAMETERS GENERATED BY TETRA (con't)

RADIAL BUILD (m)

CASE	15.4	15.5
------	------	------

(INBOARD)

INNER RAD OF OH COIL	1.695	1.697
THICKNESS OF OH COIL	.305	.303
THICKNESS OF TF COIL	.750	.750
THICKNESS OF CRYOSTAT	.050	.050
GAP	.020	.020
THICKNESS OF SHIELD	.725	.725
GAP	.020	.020
THICKNESS OF FW	.035	.035
PLASMA SCRAPE OFF	.200	.200
PLASMA RADIUS	2.000	2.000

(OUTBOARD)

PLASMA RADIUS	2.000	2.000
PLASMA SCRAPE OFF	.100	.100
THICKNESS OF FW	.035	.035
GAP	.020	.020
THICKNESS OF SHIELD	.800	.800
GAP	1.341	1.341
THICKNESS OF CRYOSTAT	.050	.050
THICKNESS OF TF COIL	.750	.750

TOKAMAK SYSTEMS CODE
RESULTS FOR ITER

PRESENTED BY DR BLACKFIELD LLNL/TRW

AUGUST 31, 1988

- IMPROVED THE NEUTRAL BEAM CURRENT DRIVE MODEL - DeVoto, Fenstermacher
- ADDED SYNCH. RADIATION MODEL - Fenstermacher
- ADDED BOOTSTRAP SLAWING - Fenstermacher, DeVoto
- BENCHMARK WITH MUMAK - Fenstermacher, Campbell
- BENCHMARK NEUTRAL BEAM MODEL - DeVoto
- EXAMINED VARIOUS IGNITION AND CD DESIGNS - Perkins
- MERGE WITH TETRA - FEDC

①

IGNITION PERFORMANCE COMPARISON OF POSSIBLE ITER DESIGN POINTSI: 20 & 15MA BASELINES

	20 MA Baseline (Cost = 1.00)			15 MA ^{option} (Cost = 1.02)		
$B_{max}(T)$	11.0	<-	<-	11.0	<-	<-
$a_{95}(m)$	1.10	0.90	<-	1.10	0.90	<-
$a_c(m)$	1.10	<-	<-	1.00	<-	<-

IGNITED/INDUCTIVE:

$R(m)$	5.00	6.00	<-	5.25	5.45	<-
$a(m)$	2.00	2.40	<-	1.62	2.02	<-
A	2.90	2.50	<-	3.25	2.70	<-
$I(MA)$	20.0	26.9	30.0	15.0	21.7	24.2
q_{ps1}	3.20	3.39	3.04	3.11	3.28	2.94
$B_0(T)$	5.10	4.03	<-	5.31	5.12	<-
$t_{BURN}(s)$	1012	597	268	1016	626	357
Q	1.80-2.50	1.52-2.15	1.36-1.92	2.08-2.50	1.67-2.36	1.49-2.11
$n_0(10^{20}m^{-3})$	1.07-1.48	0.98-1.38	0.98-1.38	1.19-1.44	1.07-1.51	1.07-1.51
$T_n(keV)$	10.0	<-	<-	10.0	<-	<-
$P_{fus}(MW)$	999-1927	1242-2483	1242-2475	738-1067	952-1904	952-1903
$P_n(MW/m^2)$	0.98-1.93	1.00-2.00	1.00-2.00	0.98-1.43	1.00-2.00	1.00-2.00
H_G	1.83-1.78	1.67-1.62	1.50-1.46	2.08-2.04	1.86-1.80	1.68-1.61
H_J	1.93-1.60	1.68-1.27	1.54-1.25	2.37-2.18	1.89-1.55	1.64-1.52
H_{10}	1.88-1.76	1.73-1.61	1.63-1.52	2.10-2.01	1.86-1.75	1.78-1.65
H_{RL}	1.01-0.79	0.72-0.54	0.64-0.49	1.42-1.26	0.96-0.73	0.87-0.66

5.8m ENG BASELINE RADIAL BUILD

(2)

IGNITION PERFORMANCE COMPARISON OF POSSIBLE ITER DESIGN POINTSII: INTERMEDIATE CURRENT AND HIGHER ASPECT RATIO

	Intermediate Current (Cost = 0.91)			Option	Higher Aspect Ratio (Cost = 0.91)			Option
$B_{\text{edge}}(T)$	11.0	<	<		11.0	<	<	
$a_{95}(m)$	2.10	0.90	<		1.10	0.90	<	
$a_c(m)$	1.02	<	<		1.00	<	<	

IGNITED/INDUCTIVE:

$R(m)$	5.53	5.73	<		5.80	6.05	<	
$a(m)$	1.81	2.21	<		1.76	2.16	<	
A	3.06	2.60	<		3.30	2.78	<	
$I(MA)$	17.5	24.3	27.1		16.9	23.8	26.5	
q_{ps1}	3.15	3.33	2.99		3.10	3.25	2.91	
$B_0(T)$	5.22	5.03	<		5.58	5.39	<	
$t_{\text{burn}}(s)$	1025	621	322		1666	1295	928	
q	1.03-2.50	1.58-2.23	1.41-2.00		1.84-2.50	1.49-2.11	1.33-1.89	
$q_e(10^{20}m^{-3})$	1.13-1.47	1.02-1.44	1.02-1.44		1.15-1.56	1.03-1.46	1.03-1.46	
$T_n(keV)$	10.0	<	<		10.0	<	<	
$P_{\text{fus}}(MW)$	874-1474	1092-2185	1092-2185		893-1647	1119-2239	1119-2239	
$P_n(MW/m^2)$	1.00-1.69	1.00-2.00	1.00-2.00		1.00-1.84	1.00-2.00	1.00-2.00	
H_G	1.94-1.89	1.75-1.70	1.58-1.53		1.82-1.77	1.83-1.59	1.47-1.43	
H_J	2.13-1.86	1.73-1.40	1.68-1.37		2.14-1.84	1.74-1.42	1.69-1.39	
H_{10}	1.97-1.86	1.80-1.67	1.69-1.57		1.82-1.71	1.66-1.54	1.56-1.45	
H_{RL}	1.18-0.98	0.82-0.63	0.75-0.57		1.20-0.97	0.83-0.63	0.75-0.57	

IGNITION PERFORMANCE COMPARISON OF POSSIBLE ITER DESIGN POINTS

III: BASELINE SUMMARY TABLE

Baseline at 5.8m

Cost	1.00	0.82	0.91	0.93
$B_{\text{trans}}(T)$	11.0	-	-	-
$d_{BS}(m)$	1.10	-	-	-
$d_c(m)$	1.10	1.00	1.02	1.00

IGNITED/INDUCTIVE:

$R(m)$	5.80	5.25	5.53	5.80
$a(m)$	2.00	1.82	1.81	1.75
A	2.90	3.25	3.06	3.30
$I(MA)$	88.0	85.8	87.3	88.9
q_{ps1}	3.20	3.11	3.15	3.10
$\theta_0(T)$	5.10	5.31	5.22	5.58
$t_{\text{BURN}}(s)$	1012	1016	1025	1066
g	1.80-2.50	2.08-2.50	1.83-2.50	1.84-2.50
$n_e(10^{20}m^{-3})$	1.07-1.48	1.19-1.44	1.13-1.47	1.19-1.56
$T_n(keV)$	10.0	-	-	-
$P_{fus}(MW)$	999-1927	738-1067	874-1474	893-1647
$P_n(MW/m^2)$	0.99-1.93	0.99-1.43	1.00-1.69	1.00-1.84
H_G	1.83-1.78	2.09-2.04	1.94-1.89	1.82-1.77
H_J	1.93-1.60	2.37-2.18	2.13-1.96	2.14-1.84
H_{T10}	1.88-1.76	2.10-2.01	1.87-1.86	1.82-1.71
H_{RL}	1.01-0.79	1.42-1.26	1.10-0.98	1.20-0.97

CURRENT DRIVE PERFORMANCE OF ITER VARIANTS AT 1 MW m^{-2}

$$[P_n = 1 \text{ MW m}^{-2}, I_{ts}/I = 0.3]$$

	ENG. BASELINE DEVICE $R_0 = 5.8 \text{ m}$	15 MA 5.25m DEVICE	17.5 MA 5.5m DEVICE	5.8m "MUKHOVATOV" DEVICE
$R_0(\text{m})$	5.8	5.25	5.53	5.8
$a(\text{m})$	2.0	1.62	1.81	1.77
A	2.9	3.25	3.06	3.30
I (MA)	20*	15*	17.5*	16.9*
$B_0(\text{T})$	5.1	5.31	5.22	5.58
g_{Troyon}	3*	3*	3*	3*
T_e/T_i (keV)	20.0/19.2	18.0/13.3	17.7/16.2	19.5/18.4
$n_e (10^{20} \text{ m}^{-3})$	0.603	0.824	0.692	0.657
NB power (MW)	58.1	62.3	58.4	52.9
LH power (MW)	20.9	28.9	24.4	22.0
Total power (MW)	78.9	91.3	82.7	74.9
Q	13.2	8.50	10.9	12.4
H_G	1.84	1.63	1.73	1.79
H_J	1.90	1.84	1.88	2.15
H_{T-10}	1.77	1.56	1.66	1.68
H_{R-L}	0.93	0.98	0.95	1.10

EBD with
 $I_{nv}/I \approx \alpha \sqrt{\epsilon/\beta_p} K^{0.15}$
 $I_{ts}/I = 0.41$
 $P_{CD} = 59.3 \text{ MW}$
 $Q = 15.8!$
 but $H'_c \geq 2$

* At upper bound.

APPROACHES FOR REDUCING THE SIZE OF ITER

5.8m Engineering
Baseline

	FROM	TO
• Increase elongation (and triag.)	2(0.4)	2.2(0.5)
• Reduce TF build by bucking soln.	0.70m	
• Reduce TF/OH gap -- leads inside	0.1m	0
• Reduce build between plasma and TF (VV+shield+FW+S.O.+gaps)	1.1m	1m
• Reduce Volt-sec cap. $\rightarrow$ min τ_{burn}	$\sim 1000s$	$\sim 500s^{*3}$
• Higher peak fields, $B_{max,TF}/B_{max,OH}$	11.2/12.5	
• Thinner magnet builds (higher J)	1.7m ^{#1}	
• Higher wall loads	1MWm ⁻²	
• Lower Z_{eff}	1.78	
• Lower q_{ψ} (95%)	3.2	
• Accept higher enhancement factors ^{#2}	$\sim 1/6-1/8$	
• Higher Troyon coefficient	2.5	
• Reduce cryostat gap	0.1m	
• Decrease scrapeoff	0.15m	
• Rely on bootstrap (even for ignition)	0	
• Decrease 20MA current requirement	20MA	
• Peak the density profiles	PlatHigh ($n_{20.5}$)	

#1, sum of OH and TF radial build

#2 $H \geq 2$ and use driven operation as a contingency. or rely on fact that $\sim 50\%$ of shots have $H \geq 2$.

#3 $\Delta F \sim 15-20cm$ reduction

(6)

REDUCTION OF ITER SIZE

By APPLICATION OF CHANGES IN (5)

Constraints: Same current (20 MA), same q_{ψ} (3.2)
same wall load (1 MW m^{-2}),
same max. fields ($B_{TF} \sim 11.2 \text{ T}$, $B_{OH} \sim 12.5 \text{ T}$)

	5.8m ENG. BASELINE	REDUCED RADIUS DEVICE
R(m)	5.8	5.12
a(m)	2.0	1.67
I(MA)	20	20
B _o (T)	5.10	5.22
T _e (keV)	10	10
β_{Troy}	1.25	1.67
Pfus	1043	845
Volt-sec, total	281	230
Volt-sec, burn	64	33
$\tau_{burn}(s)^*$	~ 1000 (± 300)	~ 450 (± 150)
H ₀	1.65	1.52
H ₅	1.77	1.89
H ₁₀	1.72	1.80
HAL	0.87	0.94
$\beta_{rel.}$	1.0	0.81

← ie room
for
~ 5cm
further bore
reduction
to give
~ 15 V
for burn.

* Range due to uncertainty in loop voltage
not uncertainty in available volt-secs.

SENSITIVITY OF SIZE REDUCTION TO INPUT CHANGES
SHOWN IN (5) AND (6)

	5.8 m EBD	→	5.1m MACHINE
Increase R from 2.0 to 2.2	✓	✓	✓
Decrease plasma -TF distance by 10cm		✓	✓
Decrease TF/oh gap by ~10cm		✓	✓
Reduce burn-time by ~ 500s			✓
$R_o(m)$	5.80	5.67	5.39
$\Delta R_o(m)$	0	0.13	0.28
Sensitivity $\Delta R_o/\Delta x$	-	6.5cm per 0.1ΔX	2.8cm per cm of plasma -TF build
		1.2cm per cm of gap	3cm per 100s of burn or 1cm per 1cm of bore.

CANDIDATE ITER DESIGN POINT AT 18MA, A=3.05

J. Perkins

August 29, 1988

	BASIC PERFORMANCE	EXTENDED PERFORMANCE
$B_{tmax}(T)$	11.0	<-
$d_{BS}(m)$	1.10	0.90
$d_C(m)$	1.03	<-
<u>IGNITED/INDUCTIVE:</u>		
$R(m)$	5.42	5.62
$a(m)$	1.78	2.18
λ	3.05	2.58
$I(MA)$	18.0	26.6
q_{psi}	3.00	3.00
$B_0(T)$	5.16	4.98
$t_{BURN}(s)$	805	260
g	1.88-2.50	1.45-2.05
$n_e(10^{20}m^{-3})$	1.14-1.52	1.03-1.45
$T_n(keV)$	10.0	<-
$P_{fus}(MW)$	845-1498	1058-2117
$P_n(MW/m^2)$	1.00-1.77	1.00-2.00
HG	1.90-1.84	1.63-1.57
HJ	2.14-1.85	1.71-1.40
HT10	1.97-1.85	1.74-1.62
HRL	1.17-0.95	0.77-0.58
<u>CURRENT-DRIVE</u>		
$I(MA)$	18.0	
g	3.0	
$T_e/T_i(keV)$	22.1 / 20.4	
$n_e(10^{20}m^{-3})$	0.680	
$E_{neut.beam}(MeV)$	1.45	
$P_{neut.beam}(MW)$	57.5 (51.2*)	
$P_{Lower\ hybrid}(MW)$	22.1 (18.0*)	
$P_{total}(MW)$	79.6 (69.2*)	
Q	11.0 (12.6*)	
$P_n(MW/m^2)$	1.0	
I_{BS}/I	0.3 (0.375*)	

* using self-consistent bootstrap of $f \approx 0.25$

SUMMARY OF POSSIBLE ITER DESIGN POINTS

CASE	1	2	3	4	5	6	7	8	9
Run number	32.7	32.10	36.4	33.1A	33.7A	33.8C	33.11C	33.11D	35.1G
R (m)	5.80	5.12	6.41	5.80	5.12	5.80	4.46	4.69	5.70
a (m)	2.00	1.67	2.36	2.00	1.67	2.00	1.58	1.32	1.91
A	2.90	3.07	2.72	2.90	3.07	2.90	2.83	3.57	2.99
I _p (MA)	20.0	20.0	25.6	20.0	19.9	18.6	14.9	11.4	20.0
B _t (T)	5.10	5.22	5.10	5.10	5.22	5.10	4.67	5.70	5.56
Elongation	2.20	2.42	2.20	2.20	2.42	2.20	2.20	2.20	2.20
Triangularity	0.60	0.75	0.60	0.60	0.75	0.60	0.60	0.60	0.60
q edge	3.20	3.20	3.20	3.20	3.22	3.44	3.20	3.20	3.20
P _{fus} (MW)	1042	846	1343	1042	847	938	643	575	980
Wall Load (MW/m ²)	1	1	1	1	1	0.90	1	1	1
n (x10 ²⁰ m ⁻³)	1.10	1.22	1.0/.96	.603	.613	.570	1.07	.924	1.0/.67
T _e (keV)	8.33	8.33	8.2/8.8	20.1	22.9	19.9	11.13	12.0	9.3/18.1
T _i (keV)	8.33	8.33	8.2/8.8	19.2	23.9	19.4	9.64	10.4	9.3/16.7
H factor	1.65	1.52	1.4/1.5	1.85	2.06	2.07	1.57	1.60	1.60
B _{max} TF (T)	11.2	11.2	11.3	11.2	11.2	11.2	12.0	12.0	12.0
B _{max} OH (T)	14.0	12.5	12.7	13.9	12.5	13.9	2.55	1.04	12.5
Burn time (s)	1733	504	2647	-	-	-	-	-	1841/-
Bootstrap frac	-	-	-	0.30	0.30	.413	0.30	0.30	/0.30
RF power (MW)	-	-	-	20.9	15.0	14.8	36.8	37.0	- /28.5
NE power (MW)	-	-	-	58.1	53.1	44.6	91.9	62.9	- /64.4
NE energy (MeV)	-	-	-	1.44	1.17	1.32	1.49	1.50	- /1.44
Q	-	-	-	13.2	12.4	15.8	5.00	5.75	- /10.6

COST SUMMARY OF POSSIBLE ITER DESIGN POINTS

(Cost in "BDE" Units)

CASE	1	2	3	4	5	6	7	8	9
Run number	32.7	32.10	36.4	33.1A	33.7A	33.8C	33.11C	33.11D	35.1G
Site & Bldgs.	7.78	7.06	8.65	7.78	7.06	7.78	6.61	6.31	7.63
Reactor Systems	23.22	17.56	30.47	23.22	17.56	23.00	13.97	13.29	22.24
Power Injection	6.80	6.80	6.80	10.27	8.91	7.74	16.65	12.76	15.56
Vacuum Systems	5.51	4.23	7.59	4.04	2.79	3.47	2.76	2.53	6.08
Heat Transport Sys	3.47	3.36	4.23	3.78	3.63	3.44	2.98	2.68	4.27
Fuel Handling Sys	3.78	3.47	4.15	3.74	3.44	3.74	3.29	3.17	3.70
I&C; Maint.; Misc.	6.99	6.99	6.99	6.99	6.99	6.99	6.99	6.99	6.99
Heat Reject. Sys.	1.12	0.98	1.40	1.25	1.10	1.13	1.02	0.91	1.51
Elec. Plant Equip.	0.76	0.76	0.79	0.98	0.91	0.87	1.17	0.98	1.93
TF (magnet+supply)	19.30	14.01	24.13	19.86	14.05	18.50	10.61	11.90	17.94
PF (magnet+supply)	20.01	13.52	30.93	21.41	15.14	20.92	12.84	5.51	18.39
Total Direct	100.0	79.72	127.8	104.6	82.59	98.79	73.98	67.78	107.4

BEAM AND FUSION ENGINEERING DIVISION

Mail Station L-644

Ext 2-6734

MEMORANDUM

9/7/88

To: Distribution

From: J. D. Lee

Subject: Weekly ITER Meeting - Wednesday 9/7/88

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LLNL - ITER STAFF MEETING AGENDA

SEPTEMBER 7, 1988

3:30 p.m. - Bldg. 543, AD Conference Room #2074

ITER Updates

D. T. Blackfield System Code Results and New Baseline
S. K. Ho Alpha Fraction in Thermally Stable Plasma
J. R. Miller ITER Reactor Magnets
R. C. Maninger Fusion in a New Era

ADDITIONAL ITEMS AND OPEN DISCUSSION

NOTE: Please provide a hard copy of viewgraphs to J. D.

University of California

LAWRENCE LIVERMORE
NATIONAL LABORATORY

SUMMARY OF POSSIBLE ITER DESIGN POINTS

Ignition

CASE	1	2	3	4	5
Run number	38.8	38.10	38.11c	38.12a	36.9a
R (m)	5.50	5.80	5.80	5.80	5.80
a (m)	1.80	2.20	2.20	2.20	2.20
A	3.06	2.64	2.64	2.64	2.64
Ip (MA)	18.0	24.0	24.0	30.0	22.0
Bt (T)	5.30	5.03	5.03	5.03	5.03
Elongation	2.20	2.07	2.07	2.20	2.07
q edge	3.11	2.97	2.97	2.63	3.24
Pfus (MW)	897	1092	1090	1139	1092
Wall Load (MW/m ²)	1	1	1	1	1
n (x10 ²⁰ m ⁻³)	1.17	1.06	.566	1.05	1.06
Te (keV)	8.33	8.33	20.3	8.33	8.33
Ti (keV)	8.33	8.33	20.3	8.33	8.33
H factor	1.74	1.59	2.50	1.29	1.73
Bmax TF (T)	11.4	11.4	11.4	11.4	11.4
Bmax OH (T)	12.9	13.0	13.0	13.1	13.0
inner shield (m)	.775	.675	.675	.675	.675
GOLDSTON	1.74	1.59	2.50	1.29	1.73
T-10	1.79	1.62	2.08	1.47	1.70
JAERI	1.98	1.64	2.36	1.49	1.69
REBUT-LALLIA	1.03	0.74	1.26	0.60	0.80

John Perkins
Sept 7, 1988.
Via Don Blackfield

CURRENT-DRIVE PERFORMANCE OF
THE NEW 5.5m, 18MA MACHINE
 $B_0 = 5.30T$

- US TETRA code; combination of NB+LH+bootstrap
- Modelling of 5.5m, 18MA technology phase
- Maximize Q by variation of T_e, T_i, n_e
and $g_{Troy} \leq 3, I \leq 18MA$
- Beam energy is allowed to vary $\leq 1.5 MeV$ for
maximum penetration at $R_{ten} = R_0$.
- Two bootstrap models are used: (i) fixed fraction of
 $\frac{I_{BS}}{I} = 0.2$, (ii) variable fraction: $\frac{I_{BS}}{I} = f(k, dT) \sqrt{\frac{a}{R}} (\beta_T - \beta_p)^{0.25}$
fast ion
- $P_n = 1 MW m^{-2}$

	FIXED BOOTSTRAP OF 0.3	VARIABLE BOOTSTRAP
$R(m)$	5.5	←
$a(m)$	1.8	←
A	3.056	←
$B(T)$	5.30	←
$I(MA)$	18	17.8
$P_{HS}(MW)$	89.7	←
$P_n(MW m^{-2})$	1.0	←
g_{Troy}	3.0	2.97
$T_e/T_i (keV)$	23.2/21.8	22.6/21.1
$n_e (10^{20} m^{-3})$	0.654	0.665
I/I_{BS}	0.3 (Fixed)	0.359
Neut. beams $\left\{ \begin{array}{l} \eta (A/W) \\ E_{beam} (MeV) \\ P_{NB} (MW) \end{array} \right.$	$\left\{ \begin{array}{l} 0.169 \\ 1.46 \\ 54.3 \end{array} \right.$	$\left\{ \begin{array}{l} 0.164 \\ 1.50 \\ 50.1 \end{array} \right.$
LH $\left\{ \begin{array}{l} f_{penetration} \\ \eta (A/W) \\ P_{LH} (MW) \end{array} \right.$	$\left\{ \begin{array}{l} 0.288 \\ 0.165 \\ 20.9 \end{array} \right.$	$\left\{ \begin{array}{l} 0.294 \\ 0.160 \\ 19.2 \end{array} \right.$
$P_{current-drive} (MW)$	75.1	70.3
Q_{total}	11.9	12.7
H_G	1.84	1.86
H_J	2.04	2.05
H_{T-10}	1.74	1.74
H_{R-L}	1.04	1.05

OPTIMIZED MULTI-MODAL-OPERATION DESIGNS FOR ITER^{a,b}

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ABSTRACT

The tokamak nonlinear optimization systems code TETRA has been expanded to contain the option of examining machine designs capable of "multi-modal" operations. The conceptual International Thermonuclear Experimental Reactor (ITER) is an example of a "bi-modal" operating machine design in that it must be capable of both pulse-ignited and steady-state operations. This new version of TETRA allows for the optimization of a "figure-of-merit" which may involve several operating scenarios. We obtain for ITER minimum cost designs which are capable of igniting when operated in the pulsed mode as well as have reasonable $Q > 5$ when operated in steady-state.

INTRODUCTION

The conceptual tokamak International Thermonuclear Experimental Reactor (ITER) is being designed to accommodate two modes of operation: pulsed ignition and steady-state current drive. Consequently, the TETRA tokamak nonlinear optimization systems code has been modified to optimize "multi-modal" designs. A multi-modal design is a flexible one which can accommodate many possible operations. A typical ITER bi-modal design would be able to achieve ignition with one set of physics parameters, while achieving high Q operations for current drive with a different set of physics parameters still using the same radial build and magnets. In this paper we will examine several possible ITER designs.

^aWork performed under the auspices of the U.S. Department of Energy by the Lawrence Livermore National Laboratory under contract No. W-7405-ENG-48.

^bDesigns specified in this paper for ITER are to be taken as typical examples only. They do not necessarily represent the final design choice.

^cTRW/LRL.

CODE DESCRIPTION

The bi-modal tokamak code examines configurations which are either pulse-ignited or current driven or both. In addition, a single operating scenario, either ignited or driven, can be examined under several different energy confinement scaling laws. For example, we examine possible ITER designs which could ignite under both Goldston¹ and Shimomura-Odaajima (JAERI) scalings².

In obtaining optimized bi-modal parameters, we maximize or minimize a "figure-of-merit" (FOM). This FOM is typically direct capital cost or major radius. For ignition designs we may alternatively try to maximize the ignition margin while for steady-state designs we may attempt to maximize $Q = \frac{\text{fusion power}}{\text{input power}}$.

The bi-modal systems code accomplishes this task by solving a set of coupled nonlinear equations which model various physics and engineering aspects using a bounded set of variables. When there are more variables than equations a unique solution does not exist so a nonlinear optimization package is used to examine the surface of solution space to obtain the "best" solution.

In the ignition designs, 60 MW of RF power is assumed to be required to startup the plasma. In current drive, both neutral beams and lower hybrid are used to drive the plasma current and these systems are also assumed to be used in startup.

In this paper we present 9 possible ITER design points. The first three are pure ignition cases, the next five pure current drive, while the last one is for a combined ignition current-drive machine. Table I shows the set of physics and engineering equations used in these cases. Table II lists the corresponding bounded set of variables used. Since there are more variables than equations, the additional ones are optimization variables which are used by the nonlinear optimization package to find an "optimal" solution.

Table 1. Equations Used in Running Bi-Modal Tokamak Systems Code

DESCRIPTION	CASE								
	1	2	3	4	5	6	7	8	9
Check on radial build	N	Y	Y	N	N	N	Y	Y	Y
Limit on OH stress	Y	Y	Y	Y	Y	Y	Y	Y	Y
Limit on TF stress	Y	Y	Y	Y	Y	Y	Y	Y	Y
B at TF coil consistency		Y	Y	Y	Y	Y	Y	Y	Y
Limit on OH current density	Y	Y	Y	Y	Y	Y	Y	Y	Y
Limit on TF current density	Y	Y	Y	Y	Y	Y	Y	Y	Y
Limit on B at TF coil	Y	Y	Y	Y	Y	Y	Y	Y	Y
Limit on TF cond. stabl. marg.	Y	Y	Y	Y	Y	Y	Y	Y	Y
Limit on OH cond. stabl. marg.	Y	Y	Y	Y	Y	Y	Y	Y	Y
TF coil quench limit	Y	Y	Y	Y	Y	Y	Y	Y	Y
Limit on B at PF coils	Y	Y	Y	Y	Y	Y	Y	Y	Y
Limit on TF conductor area	Y	Y	Y	Y	Y	Y	Y	Y	Y

(NOTE : The following equations may be solved for each mode; for bi-modal studies: 1 indicates use in mode 1; 2 use in mode 2; 1, 2 use in both modes!)

Pooidal beta consistency	1	1	1,2	1	1	1	1	1	1,2
Hot beam ion density	N	N	N	1	1	1	1	1	2
Total beta consistency	1	1	1,2	1	1	1	1	1	1,2
Total power balance	1	1	1,2	N	N	N	N	N	1
Ion power balance	N	N	N	1	1	1	1	1	2
Electron power balance	N	N	N	1	1	1	1	1	2
Limit on electron density	N	N	N	N	N	N	N	N	N
Limit on beta	1	1	1,2	1	1	1	1	1	1,2
V-S requirements	1	1	1,2	N	N	N	N	N	1
Limit on wall loading	1	1	1,2	1	1	1	1	1	1,2
Limit on plasma current	N	1	1,2	1	1	1	1	1	1,2
Limit on Q	N	N	N	1	1	1	1	1	2
Limit on divertor heat load	N	N	N	N	N	N	N	N	N
Check on neutral beam energy	N	N	N	N	2	2	2	2	2

PURE IGNITION CASES

Case 1 is the August 1988 ITER reference design with the key design parameters given in Table III and the cost breakdown in Table IV. Due to uncertainties in the costing algorithms we have listed the direct cost as 100 units. In comparing costs we will take ratios with this design and present costs as units.

Case 1, a 5.8 m machine, requires 20 MA of plasma current, wall loading of 1 MW/m², on axis toroidal field of 5.1 T, and an enhancement to Goldston scaling of 1.65 indicating partial H-mode operation (full H-mode being H = 2.0). Since the radial build has been previously determined by the ITER engineering group, the radial build equation was not used and the radial build variables are fixed.

For case 2, the shield has been reduced by 10 cm, the gap between the toroidal field (TF) and ohmic heating solenoid (OH) reduced 10 cm, the required turn time decreased by 500 s, and the plasma elongation and triangularity, listed in Table III, increased. This more aggressive engineering and physics approach lead to a reduction of the major radius from 5.8 m to 5.12 m, a reduction of the required H needed for

ignition under Goldston scaling, from 1.65 to 1.52 and an overall cost reduction over 20%. In both of these runs, the plasma current was fixed at 20 MA and a density-weighted electron temperature of 10 keV (corresponding to a volume averaged value of 8.33 keV). As shown in Table IV, the main cost savings appear in the reactor systems and the TF and poloidal field (PF) magnets.

Case 3 is the minimum size machine (R = 6.41 m) which ignites under both JAERI and Goldston scaling with H < 1.5. The first number in the rows for density, temperatures, and H corresponds to operation under Goldston scaling while the second number is for operation assuming JAERI scaling. Comparing cases 1 and 3 we see that JAERI scaling and the decrease in enhancement factor produces a higher current and larger machine. The cost of this machine would be over 25% higher than case 1. However, this machine should have a higher probability of igniting than the August 1988 reference design.

PURE CURRENT DRIVE DESIGNS

Cases 4-7 are pure current drive cases where the FOM is maximum Q. Case 4 is the

Table 2. Optimization Variables Used in Bi-Modal Tokamak Systems Code

VARIABLE DESCRIPTION	CASES							
	1	2	3	4	5	6	7	8
TF conduit case thickness	Y	Y	Y	Y	Y	Y	Y	Y
Outer TF coil case thickness	Y	Y	Y	Y	Y	Y	Y	Y
TF coil void fraction	Y	Y	Y	Y	Y	Y	Y	Y
Plasma aspect ratio	N	Y	Y	N	N	N	Y	Y
Plasma major radius	N	N	Y	N	N	N	N	Y
OH solenoid thickness	N	N	Y	N	N	N	Y	Y
OH bore radius	N	N	Y	N	N	N	Y	Y
(J per turn in TF)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
(OH stress)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
(TF stress)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
(TF inner leg thickness)	N	N	N	N	N	N	N	N
(J per turn in OH)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
(B at TF coil)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
OH conduit steel casing thickness	Y	Y	Y	Y	Y	Y	Y	Y
OH coil void fraction	Y	Y	Y	Y	Y	Y	Y	Y
OH coil copper fraction	Y	Y	Y	Y	Y	Y	Y	Y
TF coil copper fraction	Y	Y	Y	Y	Y	Y	Y	Y
(TF stability margin)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
(OH stability margin)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
(TF quench pressure J)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
(B at PF coils)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y
B_{max}	Y	Y	Y	Y	Y	Y	Y	Y
TF conductor area	Y	Y	Y	Y	Y	Y	Y	Y
(TF conductor area)/(allowable)	Y	Y	Y	Y	Y	Y	Y	Y

(NOTE: following variables may be mode dependent.)

Toroidal B field on axis	N	1	N	1	1	1	1,2	1	1
Poloidal beta	1	1	1,2	1	1	1	1,2	1	1
Ignition margin	1	1	N	N	N	N	N	N	N
(Hot beam density)/(n _e)	N	N	N	1	1	1	2	1	1
(Ti/Te)	N	N	N	1	1	1	2	1	1
Volume average T _e	N	N	1,2	1	1	1	1,2	1	1
Plasma beta	1	1	1,2	1	1	1	1,2	1	1
Volume average n	1	1	1,2	1	1	1	1,2	1	1
(Plasma beta)/(allowable)	1	1	1,2	1	1	1	1,2	1	1
(Electron density)/(allowable)	1	1	1,2	1	1	1	1,2	1	1
(Average wall load)/(allowable)	1	1	1,2	1	1	1	1,2	1	1
(V-S required)/(V-S capability)	1	1	1,2	1	1	1	2	1	1
(OH current at end of flat-top)	1	1	1,2	1	1	1	1,2	1	1
(I/Q)	1	N	N	N	1	1	2	1	1
Transport scaling H factor	N	N	1,2	1	1	1	1,2	1	1
(Plasma current)/(15 MA)	N	1	1,2	1	1	1	1,2	1	1
Plasma burn time	N	N	N	N	N	N	N	N	N
(Avg. F _{divertor})/(allowable)	1	1	1,2	1	1	1	1,2	1	1
q _{edge}	N	N	N	1	1	1	1,2	1	1
(Neutral beam energy)/(allowable)	N	N	N	1	1	1	2	1	1
(Beam path tangency radius)/(R)	N	N	N	1	1	1	2	1	1

current drive equivalent of case 1 since this has the same radial build. To obtain a maximum Q of 11.2, an enhancement to the L-mode Goldston scaling of 1.85 is required. The bootstrap fraction in this case is fixed at 30%.

The current drive design in case 5 has the same radial build as the 5.12 m ignition design of case 2. A maximum Q of 12.4 is obtained with an H of 2.06. A smaller H would yield a lower Q. Both cases 4 and 5 are slightly higher in

cost than the corresponding ignition cases with the major cost increase in the power injection system, specifically the inclusion of neutral beams which are not required in the ignition cases. Case 6 is similar to case 4 except the bootstrap fraction scales with poloidal beta. Comparing cases 4 and 6 we see that with the higher bootstrap fraction of 41.3%, Q becomes

Table 3. Summary of Possible ITER Design Points

CASE	1	2	3	4	5	6	7	8	9
R (m)	5.80	5.12	6.41	5.80	5.12	5.80	4.46	4.69	5.70
a (m)	2.00	1.67	2.36	2.00	1.67	2.00	1.58	1.32	1.91
A	2.90	3.07	2.72	2.90	3.07	2.90	2.83	3.57	2.99
I (MA)	20.0	10.0	25.6	20.0	19.9	18.6	14.9	11.4	20.0
I_p (T)	5.10	5.22	5.10	5.10	5.22	5.10	4.67	5.70	5.56
Elongation	2.20	2.42	2.20	2.20	2.42	2.20	2.20	2.20	2.20
Triangularity	0.60	0.75	0.60	0.60	0.75	0.60	0.60	0.60	0.60
q_{edge}	3.20	3.20	3.20	3.20	3.22	3.44	3.20	3.20	3.20
W_{wall} (MW)	1042	846	1343	1042	847	938	643	575	980
$n_{e,wall}$ load (MW/m ²)	1	1	1	1	1	0.90	1	1	1
n_e ($\times 10^{20} m^{-3}$)	1.10	1.22	1.0/.96	.603	.613	.570	1.07	.924	1.0/.67
T_e (keV)	8.33	8.33	8.2/8.8	20.1	22.9	19.9	11.13	12.0	9.1/18.1
T_i (keV)	8.33	8.33	8.2/8.8	19.2	23.9	19.4	9.64	10.4	9.3/16.7
H factor	1.65	1.52	1.4/1.5	1.85	2.06	2.07	1.57	1.60	1.60
β_{max} TF (T)	11.2	11.2	11.3	11.2	11.2	11.2	12.0	12.0	12.0
β_{max} OH (T)	14.0	12.5	12.7	13.9	12.5	13.9	2.55	1.04	12.5
β_{max} time (s)	1733	504	2647	ss	ss	ss	ss	ss	1841/ss
Bootstrap fraction	-	-	-	0.30	0.30	.413	0.30	0.30	10.30
RF power (MW)	-	-	-	20.9	15.0	14.8	36.8	37.0	- /28.5
NB power (MW)	-	-	-	58.1	53.1	44.6	91.9	62.9	- /64.4
NB energy (MeV)	-	-	-	1.44	1.17	1.32	1.49	1.50	- /1.44
Q	-	-	-	13.2	12.4	15.8	5.00	5.75	- /10.6

ss = steady state

larger, up to 15.8 while the plasma current decreases slightly to 18.6 MA. In the current drive runs we allow the plasma current to be 20 MA or less and we also allow the wall loading to drop to 0.9 MW/m² for case 6. Because case 6 has a higher bootstrap fraction, current drive (CD) heating requirements are lower so the total cost is lower than the design point with fixed bootstrap fraction and even slightly lower than the ignition design point of case 1.

Case 6 is the smallest major radius CD design which has a Q of 5 while Case 7 is the CD design with lowest cost. In both of these cases, the plasma current was less than 20 MA while the wall loading was fixed at 1 MW/m² and the bootstrap fraction fixed at 30%. Although the minimum cost machine is larger (4.69 m compared to 4.46 m), the Q is higher (5.75) and the neutral beam power lower (63 MW compared to 92 MW). In addition, the plasma current is much lower (11.4 MA compared to 14.9 MA). Therefore, the power injection and PF systems are less expensive than the minimum size machine.

COMBINED IGNITION AND CURRENT DRIVE

Finally, case 9 is a bi-modal ignition, current drive design. Using Goldston scaling we attempted to find the machine with the highest Q in the current drive mode which would ignite under Goldston scaling with an H of 1.6 or less for each mode. This smaller machine (5.7 m) was only 7% more expensive than the reference design of case 1.

Although the power injection system is more than twice as expensive as case 12, the reactor systems, TF and PF magnets, are all less expensive due to the smaller size and lower fusion power. For a Q around 10, the H needed for ignition is similar to the value required by power balance in the CD scenario. For Q's below 10, the required H factor for steady state operation is less than that needed for ignition.

To keep the wall loading fixed at 1 MW/m² we find that when operating ITER in the CD regime, the plasma density should be decreased by one third while the electron temperature is doubled to obtain a maximum Q of 10. The cost of this tokamak is slightly higher than the baseline design (7% higher) due to the neutral beam current drive system. However, this machine is considerably less expensive than the 6.4 m machine which used the more conservative physics assumption that H be 1.5 or less for both Goldston and JAERI scaling. One could argue that if plasma performance rather than ignition itself is the overriding concern in ITER, it might be more cost effective to design a flexible bi-modal ignition and current drive machine rather than a machine that must reach ignition under various scaling assumptions. Instead of increasing the size to accommodate possible adverse physics scaling laws, one may design the machine to accommodate current drive systems so that high Q operation may be achieved even when the plasma does not ignite.

Table 4. Cost Summary of Possible ITER Design Points

CASE	(Cost in "Units")								
	1	2	3	4	5	6	7	8	9
Site & buildings	7.78	7.06	8.65	7.78	7.06	7.78	6.61	6.31	7.63
Reactor systems	23.22	17.56	30.47	23.22	17.56	23.00	13.97	13.29	22.24
Power injection	6.80	6.80	6.80	10.27	8.91	7.74	18.65	12.76	13.56
Vacuum systems	5.91	4.23	7.59	4.04	2.79	3.47	2.76	2.53	6.08
Heat transport system	3.47	3.36	4.23	3.78	3.63	3.44	2.98	2.68	4.27
Fuel handling system	3.78	3.47	4.15	3.74	3.44	3.74	3.29	3.17	3.70
I&C; Maintenance; Misc.	6.99	6.99	6.99	6.99	6.99	6.99	6.99	6.99	6.99
Heat rejection system	1.12	0.98	1.40	1.25	1.10	1.13	1.02	0.91	1.51
Electrical plant equip.	0.76	0.76	0.79	0.98	0.91	0.87	1.17	0.98	1.93
TF (magnet+supply)	19.30	14.01	24.13	19.86	14.05	18.50	10.61	11.90	17.94
PF (magnet+supply)	20.01	13.52	30.93	21.41	15.14	20.92	12.84	5.51	18.39
Total Direct	100.0	79.72	127.8	104.6	82.59	98.79	73.98	67.78	107.4

CONCLUSIONS

The TETRA tokamak systems code has been modified to include multi-modal machine designs. A 5.8 m design has been examined as well as smaller ignition machines which use more aggressive engineering and physics assumptions. With these assumptions, cost savings up to 20% may be realized. In addition, for slightly higher costs these same machines may also be operated in steady-state using a combination of RF and neutral beams to drive the current.

Finally, a combined ignition steady-state machine may be more cost effective than a larger machine dedicated to achieving ignition if plasma performance, rather than the requirement of ignition, is the overriding concern. A tokamak which may ignite under Goldston scaling and operate with a high Q when current driven is relatively cheaper than a larger machine which may ignite under several different energy transport scaling assumptions but is not designed to accommodate the steady-state option. A bi-modal ignition/current drive experiment may cost as much as 30% more than a machine dedicated to high Q operation.

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D Lower Hybrid Heating and Current Drive

A SURVEY OF THE CURRENT EXPERIMENTAL DATABASE
FOR LOWER HYBRID CURRENT-DRIVE AND HEATING

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February 10, 1988

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A SURVEY OF THE CURRENT EXPERIMENTAL DATABASE
FOR LOWER HYBRID CURRENT-DRIVE AND HEATING

D.T. Blackfield LLNL/TRW

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ABSTRACT

The proposed ITER design may rely heavily on Lower Hybrid waves to provide heating, current drive, MHD stabilization through current profile modification and transformer recharging. This paper presents a detailed survey of recent LH experimental results from PLT, Alcator C, ASDEX, Petula-8, FT and JT-60. Current drive and heating efficiencies are given as well as regimes where sawteeth and $m = 1$ and 2 oscillations are stabilized. In addition, in ASDEX and JT-60, LH waves in combination with neutral beams (a possible ITER scenario) experiments are examined. Finally, the current drive efficiency for ITER is obtained by extrapolating from the LHCD database. Assuming 12 MW of LH power, approximately 4.5 - 5.6 MA of current could be driven in ITER. However, the high density ($\bar{n}_e = 8 \times 10^{19} \text{ m}^{-3}$) and high temperature ($T_e = 21 \text{ keV}$) will preclude wave penetration to the center. Assuming a narrow $N_{||}$ spectrum ($1.2 \lesssim N_{||} \lesssim 2$) the LH waves should be absorbed within the outer half of the plasma.

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1.0 INTRODUCTION

There are several operational scenarios envisioned for the proposed ITER design in which Lower Hybrid (LH) waves would be used. As outlined by Perkins [See Appendix 1], LH waves may be used to assist in current rampup, prolong the burn phase, assist in maintaining MHD equilibrium and help prevent disruptions. In order to assess the possibility of LH waves achieving these worthwhile goals in ITER, this paper will examine the experimental data base upon which LH heating (LHH) and current drive (LHCD) performance may be extrapolated to the reactor regime.

LH waves have been successfully launched, heated and driven current in several tokamaks. In addition, LH waves have also assisted in maintaining MHD equilibrium and prevented disruptions from sawteeth, $m = 1$ and $m = 2$ oscillations through current profile modification. In this paper, the LH wave experimental data base is surveyed for heating, current drive and disruption stabilization. Experiments on PLT at Princeton, Alcator C at MIT, ASDEX at Garching, JT-60 at JAERI, Petula-B at Grenoble, and FT at Frascati will be examined in detail. These experiments will be divided into the general categories of heating, current drive, and current profile and disruption control. In addition the design and phasing of the launching antennas will be examined. Finally, the possible implications to the design and performance of ITER will be discussed.

Several sources were used in performing this study. LH data in papers presented at the 6th and 7th Topical Conferences on RF Plasma Heating and at the 11th International Conference on Plasma Physics and Controlled Nuclear Fusion Research are used. In addition, PPPL and MIT laboratory reports from 1984 to the present on LH experiments are used. Finally articles on LH experimental results in Nuclear Fusion from 1984 to the present are also examined. Although this short list is far from exhaustive, it is believed that most of the recent LH experimental results are covered by these sources.

2.0 LH EXPERIMENTS PRIOR TO 1985

This paper focuses on the more recent (after 1985) LHH, LHCD and sawteeth suppression experiments. A survey of LHH experiments was presented by M. Porkolab at the Course and Workshop on Application of RF Waves to Tokamak Plasmas in 1985 [1]. A survey of both LHH and LHCD experiments prior to 1985 was again written by M. Porkolab as a chapter in the book Wave Heating and Current Drive in Plasmas [2]. In these papers the experimental evidence of bulk ion and electron heating, ion and electron tail formation and current drive are examined.

2.1 LH Heating Experiments

For LH waves to penetrate into the plasma from the low field edge [1],

$$\frac{\omega_{pe}^2}{\omega_{ce}^2} \sim 0.1, \quad N_{||} \gtrsim 1.2 \quad \text{where} \quad N_{||} = \frac{ck_{||}}{\omega} \quad (1)$$

and for the wave to reach the plasma interior

$$\frac{\omega_{pe}^2}{\omega_{ce}^2} \sim 1.0, \quad N_{||} \gtrsim 1.5 \quad (2)$$

On the other hand, waves with large $N_{||}$ may undergo electron Landau damping when

$$\omega/k_{||} \lesssim 3v_{te} \quad \text{where} \quad v_{te} \sim \left(\frac{2T_e}{M_e}\right)^{1/2} \quad (3)$$

Based on quasi-linear Fokker-Planck calculations, the factor of 3 in Eq. (3) is slightly lower, so LH waves will be heavily damped when

$$\frac{7.0}{[T_e(\text{keV})]^{1/2}} \gtrsim N_{||} \quad (4)$$

Together Eqs. (2) and (4) specify that a window exists in $N_{||}$ space where LH waves may propagate. Equation (4) indicates that for ITER, LH waves will be damped on the electrons well before they reach the center, thereby being unable to drive current there. In addition, because of this theoretical wave launching window, it is important in obtaining high efficiency current drive to launch LH waves with most of the power in this window.

A further complication to the $N_{||}$ picture is that LH waves in toroidal plasmas suffer upshifts and downshifts in $N_{||}$ as they propagate through the plasma and reflect at the walls. Shifts in $N_{||}$ are also predicted as the LH waves are reflected from density gradients and local density fluctuations which further muddles the predicted wave spectrum. In fact on performing LHCD experiments, it was found that some "filling in" of the $N_{||}$ spectrum must occur in order to relate the observed current drive efficiencies with the theoretical predictions.

2.2 Ion Heating Experiments

Finite ion and electron temperature effects cause the LH wave to mode convert into an ion plasma wave when $\omega \sim \omega_{LH}$. Strong perpendicular ion Landau damping should occur when

$$\omega \sim 3^{1/2} k_{\perp} v_{ti} \quad (5)$$

Several LHH experiments in the 1970's examined this ion heating regime at low power levels (100 - 200 kW). In all of these experiments hot ion tails were observed. In addition bulk ion temperature rises of 100 - 200 eV were seen. Two of the experiments in which the highest ion heating were obtained were in the JFT-2 and Petula tokamaks.

In JFT-2, a bulk ion temperature rise of 500 eV was measured for 200 kW of LH power launched into a $\bar{n}_e \sim 1 \times 10^{19} \text{ m}^{-3}$ plasma. A maximum heating rate of $\eta_{\text{HEAT}} = 2.5 (\times 10^{19} \text{ eV/m}^3/\text{kW})$ was calculated where

$$\eta_{\text{HEAT}} = \frac{\Delta[\sum_j \bar{n}_j T_j(0)]}{P_{\text{RF}}} \quad (6)$$

where $\bar{n}_j$ is the line-averaged density and $T_j(0)$ the central ion temperature. In Petula an RF frequency of 1.3 GHz for a $B \sim 30$ kG and density of $\bar{n}_e \sim 6 \times 10^{19} \text{ m}^{-3}$ has produced $\eta_{\text{HEAT}} \sim 3 \times 10^{19} \text{ eV/m}^3/\text{kW}$.

However in experiments performed on PLT, Alcator C, FT and ASDEX efficiencies well below $3 \times 10^{19} \text{ eV/m}^3/\text{kW}$ were seen. The ion heating appears to be dominated by hot ion tail formation near the plasma surface which are poorly confined. This tail formation may be a product of parametric decay instabilities.

As M. Porkolab summarizes in his 1985 article [1], ion heating results are machine dependent and there is poor correlation between theory and experiment. It appears that LH waves launched in the ion heating regime will undergo damping at the plasma edge, producing edge heating in large tokamaks with flat density profiles. Besides parametric instabilities and ion tail formation near the plasma edge, collisional absorption near the plasma edge will also contribute to wave damping at the plasma edge. It is reasonable to assume that ITER will not be a good candidate for LH ion heating.

2.3 Electron Heating Experiments

For frequency $\omega \geq 2\omega_{\text{LH}}(0)$ LH waves may undergo electron Landau damping when $N_{\parallel}$ satisfies Eq. (4). Electron heating has been observed in most of the LH experiments often together with current drive when the antenna is phased $\Delta\phi = 90^\circ$. Table 1A, from Ref. [1], is a summary of the electron heating experiments prior to 1985 in several tokamaks. In nearly all of the experiments, the heating efficiency is much higher than that obtained in LH ion heating experiments.

To summarize these early experiments, for $\omega/\omega_{\text{LH}} \geq 2$, direct electron heating is seen while the ions are heated through ion-electron collisions. An upshift in the $N_{\parallel}$ spectrum must occur because the spectrum launched by the antennas are too low to obtain efficient heating using quasi-linear theory. Heating rates vary from $\eta = (3 - 4) \times 10^{19} \text{ eV/m}^3/\text{kW}$ in low density tokamaks (ASDEX and PLT) to $\eta = (10 - 26) \times 10^{19} \text{ eV/m}^3/\text{kW}$ in high density tokamaks (Alcator C and FT). For densities where $\omega/\omega_{\text{LH}}(0) \lesssim 2$, ion tails are generated usually accompanied by parametric spectra in the scrapeoff region. These tails are poorly confined and little bulk heating occurs.

TABLE 1A

RECENT LOWER HYBRID HEATING RESULTS

DEVICE	f(GHz)	$\bar{n}_e (10^{19} \text{ m}^{-3})$	B(T)	$\Delta T_e (\text{keV})$	$\Delta T_i (\text{keV})$	P(kW)	η	COMMENT
Alcator C	4.5	14	9.0	1.0	0.8	850	22-26	SiC Limiter
		15	9.0	0.27	0.10	500	11	Mo Limiter
		15	5.5	0.50	0.25	700	15	Mo Limiter + Carbonized Walls
ASDEX	1.3	2.0	2.2	0.50	0.10	400	3	Electron Htg.
FT	2.45	4.0	8.0	1.0	0.4	280	~20	Electron Htg.
PLT	2.45	0.8	2.9	2.0	---	450	4	Electron Htg.
JFT-2	0.75	2.0	1.0	---	0.3	200	3	Ion Heating
		1.0	1.0	---	0.5	200	2.5	Ion Heating
PETULA-B	1.3	6.0	3.0	---	0.5	700	4	Ion Heating
PLT	0.8	1.9	2.7	---	0.25	350	1.3	Ion Heating

$$\eta = \frac{\sum_j \Delta \bar{n}_j (10^{19} \text{ m}^{-3}) T_j (0) (\text{eV})}{P_{RF} (\text{kW})}$$

2.4 Current-Drive Experiments

Prior to 1985, LHCD experiments have been performed in several tokamaks throughout the world. Table 1B from Ref. [2] summarizes the parameters from these experiments. This density limit is the density above which no current drive has been observed. Prior to 1985, in Alcator C, with $B = 8T$, $T_e(0) \sim 1.5$ keV, $T_i(0) \sim 0.7$ keV, for LH power of 250-1000 kW, 140 kA of current was driven in a density range from $1 \times 10^{19} \text{ m}^{-3}$ to $7 \times 10^{19} \text{ m}^{-3}$.

3.0 PLT LH EXPERIMENTS

The first LHCD experiments which will be examined in some detail were performed on the PLT tokamak from 1983 - present. The earlier experiments were performed with an RF frequency of 800 MHz. Later experiments used different transmitters which operated at 2.45 GHz. PLT parameters are given in Table 2A.

3.1 PLT Current-Drive Experiments (800 MHz operation)

In the earlier PLT current drive experiments, up to 500 kW of LH power drove steady-state plasma currents up to 175 kA for 3 seconds and 400 kA for 0.3 seconds with an RF frequency of 800 MHz [3]. Using 800 MHz, current could not be driven for electron densities greater than $8 \times 10^{18} \text{ m}^{-3}$.

In 1984, the same 800 MHz RF system was used to initiate and drive a plasma current [4]. In this experiment up to 200 kW of power was launched through either a narrow 2 cm waveguide grill ($N_{||} \sim 3.5$) or a wider 3.55 cm grill ($N_{||} \sim 2$). A background deuterium gas of $(1 - 2) \times 10^{-5}$ Torr was broken down by the RF forming a 10^{18} m^{-3} plasma in a 10 - 30 ms time period.

Following this initial breakdown, the waveguide array was changed from 0° phasing to either 90° or 135° and current was driven in either toroidal direction. The narrow waveguide grill could easily initiate and drive current while the broader grill required careful programming of the vertical field to maintain plasma equilibrium. The current initially rises to 30 kA at a rate of $\sim 200 \text{ kA/s}$. For the next 2 seconds the current ramps up at a slower 35

TABLE 1B

STEADY STATE CURRENT DRIVE EXPERIMENTS - 1985

DEVICE	P (MW)	f (GHz)	GRILL	B(T) (TESLA)	DENSITY LIMIT (m^{-3})	COMMENT
Alcator C	1.2	4.6	4 x 4	7 - 11	10×10^{19}	<u>Highest</u> n ($E_{dc}=0$)
ASDEX	0.8	1.3	1 x 8	1 - 2.5	1.5×10^{19}	<u>Diverted Plasma</u>
FT	0.2	2.45	2 x 2	6 - 8	3×10^{19}	
JIPP T-II-U		0.8	1 x 2	3	?	
PETULA	0.7	1.3	1 x 4	3	1.5×10^{19}	<u>Improved MHD Stability</u>
PLT	0.5	0.8	1 x 6	3	0.8×10^{19}	<u>Longest Pulse</u>
		2.45	1 x 8	3	?	($E_{dc} = 0$)
VERSATOR II	0.1	0.8	1 x 4	1 - 1.5	0.6×10^{19}	
		2.45	1 x 4	1 - 1.5	$> 1.7 \times 10^{19}$	$\omega_{pe}^2/\omega^2_e = 1$
WT-II	0.1	0.8	1 x 4	0.5	?	

TABLE 2A

PLT PARAMETERS

	800	2.45	SAWTEETH
	MHz	GHz	SUPPRESSION
R (m)	1.32	1.32	1.32
a (m)	0.4	0.4	0.4
B (kG)	20	20-32	29
$\bar{n}_e$ ($\times 10^{19} \text{ m}^{-3}$)	0.1-0.2	0.2-4	1
I_p (kA)	300-700	~ 500	500
q_L	$2 < q_L < 6$		
f (GHz)	0.8	2.45	2.45
P_{RF} (kW)	500	≤ 1000	≤ 420
f_{pulse} (s)	0.3-3		
η_{CD} ($\times 10^{19} \text{ A/m}^2/\text{W}$)	0.6-1	.14	---
η_{HEAT} ($\times 10^{19} \text{ eV/m}^3/\text{kW}$)		4	---

kA/s rate. It was concluded that over 90% of the RF power was unaccounted for. In addition, an initial increase in the vertical field of 0.01 Volt-seconds was needed to generate an initial seed of electrons with energies above 20 keV. The LH current efficiency was greater than that predicted by quasi-linear theory. Figure (1) from Ref [4] shows the time evolution of the LH driven current, vertical field intensity, line-averaged plasma density and poloidal beta plus internal inductance.

By 1985 current rampup in PLT was achieved at somewhat higher densities ($\bar{n}_e \sim 2 \times 10^{18} \text{ m}^{-3}$) with rampup efficiencies greater than 25% where $\epsilon = \dot{W}/P_{RF}$ and $\dot{W}$ is the power flowing into the poloidal field [5-6]. Figure 2 from Ref. [6] shows the plasma current driven by the LH waves for various RF powers. For $\bar{n}_e = 2.2 \times 10^{18} \text{ m}^{-3}$, and a waveguide phasing of 60° , 40 kW was required to maintain the Ohmic produced 200 kA plasma current. The current was increased by a rate up to 120 kA/s using 260 kW of RF power with an inductive rampup power of 89 kW [6]. A hot spot on the limiter producing an uncontrollable increase in plasma density limited the rampup rate. In these experiments the OH power supply was nearly clamped at various output currents. This clamping produced approximately 15 kW of drive power. In other cases, the OH supply was "open-circuited" so that no current drive power was supplied. It was found that the highest current rampup rates were obtained at the lowest densities using the fastest (smallest $N_{||}$) waves. The current drive efficiency was found to remain constant for RF powers 2 ~ 3 times that required to maintain plasma current equilibrium.

In OH current drive (OHCD), the magnetic flux flows from the outside in towards the plasma center, creating an electric field parallel to the driven current. In LHCD, the flux is created near or at the plasma center where the RF power is absorbed. This flux leaks outwards, creating an electric field anti-parallel to the driven current. This electric field will cause bulk electrons to flow opposite to the desired direction, thereby reducing the overall current drive efficiency. In addition, since LH waves also interact with superthermal electrons [1-6], this induced electric field can also produce a second anti-parallel flowing current made up of runaway electrons thereby further reducing the current drive efficiency [5]. Finally, the antenna itself will launch some power in the opposite direction [7] although a well designed antenna grill will attempt to minimize the fraction of backward

traveling power. The X-ray data obtained from a limited number of ramp-up shots indicates that backward traveling superthermal electrons may carry on the order of 10% of the forward current [5]. The bulk backward current $I_b = V/R$ where V is the axial loop voltage and R the bulk plasma resistance. The bulk plasma resistance cannot be measured directly but a simple power balance model used to estimate T_e finds that a bulk plasma current of 70 kA or about 1/3 of the forward current is produced. Therefore at the highest ramp-up rates, an induced backward flowing plasma current was produced which appeared to be 30 - 50% of the forward one.

Reference [6] found that up to 20% of the full plasma current in PLT could be reached during current rampup experiments with a rampup efficiency (ratio of increase in the magnetic field energy to RF energy used) of 20%.

3.2 PLT Current-Drive Experiments (2.45 GHz operation)

In 1985 current drive experiments were being performed in PLT at higher densities ($1 - 2 \times 10^{19} \text{ m}^{-3}$) using up to 1 MW of RF power at a higher frequency (2.45 GHz) [8]. The PLT operating regime for this higher frequency is given in Tables 2a and 2b. There were two identical antennas used in PLT; one on the midplane the other on the top. Each coupler launching up to 500 kW of power was composed of 8 waveguides ($1 \times 10 \text{ cm}^2$). With both antennas phased at 90° , the side antenna proved to be more efficient in driving current. Ray tracing calculations indicate that a top launched LH wave undergoes an up-shift in $N_{||}$, thereby producing a slower wave which is predicted to be less efficient in driving current. When the phasing of the antennas were adjusted to produce identical $N_{||}$ spectra at the center, the current drive efficiencies were the same. It also appears from ray tracing calculations that the LH waves are damped within one pass through the plasma at this higher density.

Reference [9] compares the current drive results obtained at 2.45 GHz with those obtained at 800 MHz. The current drive efficiencies obtained at these two frequencies are shown in Fig. (3) from Ref. [9]. The density limit beyond which no current could be driven was raised to $\sim 4 \times 10^{19} \text{ m}^{-3}$, approximately 6 times higher than that obtained in the earlier 800 MHz experiments. The current drive efficiency $\eta_{CD} = \frac{n_e I_p R}{P_{RF}}$ of $.14 (\times 10^{20} \text{ A/m}^2/\text{W})$ [10] is

TABLE 2B

PLT ELECTRON HEATING PARAMETERS (2.45 GHz)

$R(m)$ =	1.32
$a(m)$ =	0.39
$B(kG)$ =	31
$f(GHz)$ =	2.45
$\bar{n}_e (\times 10^{19} m^{-3})$ =	1

	LHCD (Current Drive)	LHSYM (Symmetric- Heating mode)	LHOD (anti-current)
Phasing $\Delta\phi$	90°	180°	-90°
$N_{ }$	1.8	≈ 1.8	-1.8
I_p (kA)	526	517	524
I_{ST} (kA)	618	414	320
V_{Axis} (V)	-0.10	0.13	0.4
P_{RF} (kW)	585	563	584
P_{TOT} (kW)	532	630	794
$(P_{RF} + P_{OH}) \tau_E (ms)^*$	19.3	14.9	12.6
Z_{eff}	3.4	3.4	6

* OH plasma has $\tau_E \approx 17$ ms

greater than that obtained in the 800 MHz experiments and was achieved when launching the fastest waves. The 6 element waveguides used in the 800 MHz experiments had $N_{||} = 1.55$ and 2.7 with $\eta_{CD} \sim .1$ and $\sim .06$ [10], the 8 element waveguide $N_{||} = 1.4$ [9] while the 16 element has a $N_{||} = 0.45$ [10]. Above $1 \times 10^{19} \text{ m}^{-3}$, the current drive efficiency improves as the plasma current and electron temperature increases. A plasma current of 770 kA was maintained using 680 kW of RF power at $\bar{n}_e = 0.95 \times 10^{19} \text{ m}^{-3}$ [10]. The best current drive efficiency obtained by the narrow 16-waveguide grill is approximately 80% of the theoretical value ($\eta_{CD} = 0.53$) [10].

3.3 PLT Heating Experiments (2.45 GHz operation)

Besides broadening the current profile, LH produces a central electron temperature rise and a resulting peaked temperature profile [10-12]. The electron temperature rise saturates for RF powers greater than 500 kW [10]. The central electron temperature rises from ~ 2.2 keV to ~ 5 keV [12]. Although the stored energy increases as the RF power is increased, the rate of increase is less than that obtained by Ohmic heating. As the electron density rises, the temperature rise decreases. For RF power levels comparable to Ohmic power levels, the electron energy confinement times are similar. For electron densities less than $3 \times 10^{19} \text{ m}^{-3}$, the confinement time increases with density but at one third the rate obtained from Ohmic discharges.

The best heating results were obtained from LH waves launched parallel to the electron drift velocity (current drive mode) [11-12]. Waves launched symmetrically or anti-parallel produced significantly smaller electron temperature rise. Figure (4) from Ref. [12] shows the electron temperature profile measured from Thompson scattering for 500 - 600 kW of LH power launched at various phasing from parallel (-90°) to -180° at a line average density of 10^{19} m^{-3} . The PLT parameters for this study are give in Table 2b. Although central temperatures as high as 6 keV were obtained [12] the outer temperature ($r/a > 0.5$) were similar in all phasings and to no RF.

3.4 PLT Sawteeth Oscillations and MHD Instabilities Suppression (2.45 GHz)

Besides using LH to drive current and heat particles, experiments have demonstrated that LH can modify the current profile and suppress sawteeth [13]. The PLT sawteeth suppression parameters are given in Table 2a. In the experiments discussed in Ref. [13], the plasma current and density are kept constant from $t = 400$ ms to $t \sim 900$ ms. Sawteeth oscillations appear at $t = 400$ ms and reach a quasi-steady state 100 ms later. At this time the central electron temperature is ~ 2 keV with an oscillation amplitude of $\sim 10\%$ and a period of 6 ms. These oscillations were eliminated with 40 kW of RF power. PLT has shown [9] that there is an RF power threshold which depends upon the plasma density above which sawteeth oscillations can be eliminated. Above $\bar{n}_e \sim 3 \times 10^{19} \text{ m}^{-3}$, there was not enough available RF power (1 MW) to eliminate the sawteeth oscillations (see Figure 5 from Ref. [9]).

At higher RF powers, the $m = 1$ instability can also be stabilized in PLT. With $q_L > 2.5$, LH waves suppressed both the $m = 0$ and $m = 1$ modes without increasing $m = 2$ modes or creating plasma disruptions. When operated with $q_L = 2$, strong $m = 2$ modes were present and plasma disruptions occurred in both Ohmic alone and LHCD plasmas [9].

LH waves appear to stabilize the $m = 0$ and $m = 1$ instabilities by broadening the current profile and increasing the plasma conductivity [14-15]. In PLT discharges with $q_L > 3$, $m = 2$ activity is absent but appear in discharges when $q_L < 2.5$ [14]. While $m = 1$ and $m = 2$ modes may be present during quasi-steady state current drive, $m = 2$ and $m = 3$ modes appear during current rise operation.

In PLT with a plasma current of 500 kA, the rate of decrease in the internal inductance or the rate of increase in the width of the plasma current channel is $\sim 0.1/\text{MW}$ [15]. For RF powers of 450 kW, sawtooth oscillations are suppressed with the $m = 1$ magnetic island being eliminated while the current channel radius is increased approximately 1.5 cm [15]. During the LH current drive experiments where the waves are launched from two grills with phasing -60° , the inversion surface of the sawtooth oscillations is shifted outwards, $m = 1$ island width is increase, and the internal inductance decreased. For an RF power level of 245 kW, the inversion surface shifts outwards approximately 2 cm. At higher power levels, the oscillations are suppressed, indicating

that the inversion surface is eliminated. However, $m = 1$ oscillations are still present for RF power levels less than 450 kW.

For zero power RF discharges the $m = 1$ island width is < 1 cm increasing to ~ 10 cm as the RF power levels rises to 300 kW. This indicates that the q profile is flattening in the area of the $q = 1$ surface. When the electron temperature profiles from Ohmically driven discharges are compared with those obtained from LHCD ones, the half-width at half amplitude is approximately 3 cm broader in the LHCD cases. Finally, the plasma internal inductance is lowered as the RF power increases, also indicating a current channel broadening.

In PLT and other tokamaks ICRF produces large amplitude oscillations and a similar phenomenon is expected to occur in devices like ITER when hot alphas are present. In addition to the 2.45 GHz LH systems with 0.8 MW of power, a 30 MHz, 2 MW ICRF system was also on PLT. When ICRF minority heating was present without LHCD, for 1 MW of ICRF power the sawteeth amplitude increased from 0.20 keV to 0.5 keV. The use of LH waves stabilized these sawteeth as well, with a minimum amount of power, independent of the amount of ICRF power up to 1 MW [9]. For a plasma current of 500 kA, $q_L = 3.7$, $\Delta\phi = -75^\circ$, and $\bar{n}_e \sim 2.5 \times 10^{19} \text{ m}^{-3}$, approximately 700 kW of LH power suppressed oscillations for RF powers up to 800 kW. Reference [16] also discusses using LH waves to suppress sawteeth when ICRF is present. As the plasma density increases, the current driven by LH decreases and the amount of LH RF power required to suppress the ICRF enhanced sawtooth oscillations increases. In Ohmic discharges, the sawteeth amplitude is about 10% of the central temperature while in ICRF discharges with 2 MW of power, this amplitude increases to 20%. Although this amplitude is doubled that of the Ohmic cases, the amount of LH power required to eliminate these oscillations is the same in the two cases. This occurs because the shapes of the electron temperature profiles in the Ohmic heating and ICRF heating discharges are the same. Since the current profile is usually determined by the shape of the temperature profile (when LH waves which produce current profile modification are not present), the current profiles are similar so it is expected that the amount of LH power needed to broaden the current profile should be the same. It was concluded in Ref. [16] that LHCD can be used in ICRF heated plasmas without degrading the energy confinement time or the ion heating efficiency of the ICRF. At high power

levels the operating of the LH and ICRF systems together is more complicated. The optimal edge condition required for one system is often much less desirable for the other. With careful vacuum vessel conditioning and plasma positioning, both systems can function well together [16].

4.0 LH EXPERIMENTS IN ALCATOR C

The first LHM and LHCD experiments investigating wave propagation and absorption at high densities were performed in Alcator C at MIT. The Alcator C LH system is capable of delivering up to 1 MW of RF power at a frequency of 4.6 GHz into a plasma with electron densities up to $2 \times 10^{20} \text{ m}^{-3}$ with a toroidal field as high as 110 kG.

4.1 Current-Drive Experiments

Current drive experiments using up to 1 MW of RF power have been performed on the Alcator C tokamak at MIT at high fields and densities [17-18]. The Alcator C LHCD parameters are give in Table 3 [20]. A current drive efficiency of $\eta_{CD} = \frac{n_e I_p R}{P_{RF}}$ of 0.12 ($\times 10^{20} \text{ A/m}^2/\text{W}$) was reached with $B = 10 \text{ T}$ and a lower efficiency of 0.08 at a lower field of 8 T [17]. The current is carried by suprathermal electrons in the tail of the distribution as also seen in PLT. Typically, to drive 140 kA of plasma current at $\bar{n}_e = 3 \times 10^{19} \text{ m}^{-3}$ requires 320 kW of power. The peak temperatures of both Ohmic discharges and RF driven discharges are similar as are their profiles [17]. For a discharge with 320 kW of RF power the gross energy confinement time is approximately 3 ms which is smaller than a similar OH discharge with OH power of $\sim 210 \text{ kW}$ and a confinement time of $\sim 4.5 \text{ ms}$. Figures (6a-6b) from Ref. [17] show the comparison of confinement times between OH and RF driven discharges as either the density or magnetic field is varied. Both the confinement time improves and the current drive efficiency increases as the magnetic field increases while the OH energy confinement time is relatively insensitive to the magnetic field. The ratio of $\frac{\tau_E^{ef}}{\tau_E^{OH}} \sim 0.22/P_{RF} \text{ (MW)}$. There appears to be no increase in

TABLE 3

ALCATOR C PARAMETERS

	LHCD	LHH	SAWTEETH SUPPRESSION
R(m)	0.64	0.64	0.64
a(m)	0.165	0.165	0.165
B(kG)	70-110	55-90	60
$\bar{n}_e$ ($\times 10^{19} \text{ m}^{-3}$)	3-7	≤ 20	6-12
I_p (kA)	100-200	150-375	250-325
q_L	≥ 8		4-6
Z_{eff}	1.5-2		
f (GHz)	4.6	4.6	4.6
P_{RF} (MW)	≤ 1.5		≤ 1.4
f_{pulse} (ms)	300		
η_{CD} ($\times 10^{20} \text{ A/m}^2/\text{W}$)	0.12 at 100 kG 0.08 at 80 kG	---	---
η_{HEAT} ($\times 10^{19} \text{ eV/m}^3/\text{kW}$)	---	11-26	---

the confinement time in either the OH or the RF driven discharges with increasing plasma current [17].

At low densities ($\bar{n}_e < 3 \times 10^{19} \text{ m}^{-3}$) the energy confinement time in LHCD discharges is somewhat larger than for OH ones. At higher densities, the confinement time is lower being comparable to Kaye-Goldston scaling used in neutral beam heated plasmas [19-20]. In addition, there are indications that energy confinement times are larger in discharges with OH and LH current drive than similar ones with OH alone [20]. LH may be used to replace some of the OH capability in ITER as well as increase the energy confinement time.

In current rampup experiments [18], at densities $\bar{n}_e > 1 \times 10^{19} \text{ m}^{-3}$ the rampup efficiency $\frac{P_{e1}}{P_{RF}} \sim 100\%$ initially but quickly drops to 5 - 10% after 50 ms.

4.2 LH Electron Heating Experiments

Electron heating experiments have been performed in Alcator C with parameters given in Table 3 [19]. Bulk electron heating from electron Landau damping has been observed in plasmas with $\bar{n}_e$ from 6 to $7 \times 10^{19} \text{ m}^{-3}$. $T_e(0)$ has increased 0.7 keV and $T_i(0)$ increased 0.3 keV for 1 MW of power in a plasma with $B = 5.5 \text{ T}$, $I_p = 260 \text{ kA}$ with a change in Z_{eff} less than 0.5. Comparable heating efficiencies are obtained using current drive phasing of the antenna ($\Delta\phi = 90^\circ$) as well as from pure heating mode ($\Delta\phi = 180^\circ$) [19]. Figure (7) from Ref [21] shows the increase in both $T_e(0)$ and $T_i(0)$ at various power levels for current drive (90°) and anti-current drive (-90°) phasings. Included are both shots when the sawteeth were and were not stabilized. The plasma conditions in this study were $\bar{n}_e = 7 - 8 \times 10^{19} \text{ m}^{-3}$, $I_p = 270 \text{ kA}$, $T_e^{OHMIC}(0) \approx 1.3 \text{ keV}$, $T_i^{OHMIC}(0) \approx 0.75 \text{ keV}$.

4.3 Sawteeth Stabilization Experiments

As in PLT, sawtooth oscillations have been eliminated in Alcator C using LH with plasma parameters given in Table 3 [19,21]. For $\bar{n}_e = 11 \times 10^{19} \text{ m}^{-3}$, $I_p = 250 \text{ kA}$ and $B = 6.2 \text{ T}$, sawteeth were stabilized for RF powers greater than 550 kW. They have been eliminated only when the antenna grills are operated

with current drive phasing (90°). For power levels above 850 kW, the sawteeth reappeared after 40 - 50 ms, a time comparable to the L/R time. For $P_{RF} \sim 1.3$ MW, the sawteeth amplitude is comparable to the pre-injection level [21]. To stabilize the sawteeth in Alcator C the LH must drive at least 20% of the total plasma current [21].

Unlike PLT, the sawtooth inversion radius remains fixed at $r/a \sim 0.2$ for all RF power levels until they are eliminated [19,21]. Occasionally $m = 1$ oscillations are seen in fully stabilized discharges. The sawteeth period generally increases with RF power as shown in Figure (8) from Ref. [21].

5.0 LH EXPERIMENTS IN ASDEX

Heating and current drive experiments have been performed on the ASDEX tokamak. Up to 1 MW of RF power has been launched into ASDEX at a frequency of 1.3 GHz. This lower frequency requires ASDEX to operate at lower densities ($0.5 \cdot 10^{19} \text{ m}^{-3}$) to avoid the density cutoff. In addition, up to 3.4 MW of neutral beam power has been injected into ASDEX to investigate the effects of NBI + LHCD. The ASDEX parameters are listed in Table 4.

5.1 Current-Drive Experiments

ASDEX found efficient savings (fraction of OH power replaceable by LH power ~ 1 for densities below $\bar{n}_e = 0.5 \times 10^{19} \text{ m}^{-3}$) of volt-seconds in the primary OH transformer were obtained by combining OHCD with LHCD.

When neutral beams were also included, the current drive efficiency dropped as shown in Figure (9) from Ref. [22]. It is believed that the hot beam ions may directly absorb some of the LH wave power, thereby reducing the energy available to drive current. The LH waves accelerate the beam ions up to twice the injected energy at the plasma center. As shown in Figure (9), the current drive efficiency decreases as the beam power increases. There is also a corresponding drop in the hard X-ray intensity and suprathermal ECE emission from the fast electrons when neutral beams are used [22].

TABLE 4

ASDEX PARAMETERS

	LHCD	LHH	SAWTEETH SUPPRESSION
R (m)	1.65	1.65	1.65
a (m)	0.4	0.4	0.4
B (kG)	22	22	22
$\bar{n}_e$ ($\times 10^{19} \text{ m}^{-3}$)	~ 0.5	0.6-2	~ 1.2
I_p (kA)	300	300	380
q (a)			2.75
f (GHz)	1.3	1.3	1.3
P_{RF} (kW)	≤ 1000	375	550
P_{NBI} (kW)	< 3.4		≤ 1.8
f_{pulse} (s)	≤ 1.5	≤ 1.5	≤ 1.5
η_{CD} ($\times 10^{20} \text{ A/m}^2/\text{W}$)	$\sim .06$	---	---
η_{HEAT} ($\times 10^{19} \text{ eV/m}^3/\text{kW}$)	---	3	---

5.2 Electron Heating Experiments

Strong electron heating is observed in ASDEX when LH wave power is absorbed by suprathermal electrons at plasma densities below $3 \times 10^{19} \text{ m}^{-3}$. At a density of $0.6 \times 10^{19} \text{ m}^{-3}$, $P_{\text{OH}} = 240 \text{ kW}$, $P_{\text{LH}}^{\text{abs}} = 375 \text{ kW}$ $T_e(0)$ increases from 2 keV to 3.5 keV. T_e increases in the central region ($r/a < 0.25$) and remains similar to OH discharges outside. The heating zone appears to be restricted to the $q = 1$ surface [22]. Since sawteeth are only suppressed in the LHCD mode while similar heating results appear in the LHH mode, this central heating seems to be independent of the presence of sawteeth. Using a simple power balance, it is calculated that the LH absorption efficiency decreases from 0.8 at $\bar{n}_e = 1 \times 10^{19} \text{ m}^{-3}$ to about 0.3 at $\bar{n}_e = 2 \times 10^{19} \text{ m}^{-3}$. The absorption efficiency also decreases with increasing RF power. It is thought that as the density increases, the LH waves become less accessible to the center and saturation effects may occur at higher powers. Also as the RF power increases, the energy confinement time decreases. LHH is more efficient when neutral beams are used to pre-heat the plasma.

The density limit above which no LHH occurs is believed to be caused by the RF being absorbed by ions [23]. ASDEX has observed that fast ions are produced at high densities in the outer half of the plasma [23]. Also, a hollow electron temperature profile was observed, indicating that the LH waves were being absorbed in the plasma outer region, even though calculations indicate that with the spectrum being launched, some RF power should propagate to the center.

5.3 Current Profile Modification

In LHCD discharges and those which also include neutral beams, the electron temperature and current profiles become decoupled. The current profile broadens while the electron temperature peaks. As seen in PLT and Alcator C, sawteeth can be suppressed only with LHCD phasing. It appears that LH waves with slower phase velocities are absorbed near the plasma edge. This off-axis absorption and resulting current drive leads to a broadening of the current profile. Alcator C, a high density LHCD experiment observed that

the sawteeth period increased monotonically with increasing RF power until stabilization occurred in contrast with the ASDEX and PLT experiments [19].

Sawteeth were also stabilized in neutral beam heated plasmas with LHCD phasing. To stabilize the sawteeth without neutral beams, about 50% of the LH power needed to drive the plasma current was required. With neutral beams, the power required decreases as the neutral beam power increases. Stabilizing sawtooth oscillations increases $T_e(0)$ and improves the energy confinement. In ASDEX from 1/3 - 2/3 of the LH power required to drive the current completely must be supplied to eliminate sawteeth [24]. The amount of power increases with electron density and plasma current. Sawteeth are stabilized in ASDEX when $q > 1$ throughout the entire plasma. However, in medium and high density discharges, $m = 1$ oscillations may still be present, indicating that a $q = 1$ surface lies within the plasma, even when sawteeth are eliminated.

The threshold power required to eliminate sawteeth in neutral beam heated plasmas is higher than for OH discharges. However, with high neutral beam power, sawteeth are eliminated with neutral beams alone. The time delay between the injection of LH power and the elimination of sawteeth decreases as the neutral beam power increases for constant LH power. It appears that neutral beams can flatten the current profile in the center thereby eliminating sawteeth.

LH waves often produce $m = 2$ oscillations [23]. Usually these oscillations saturate at low levels. To prevent LH waves from triggering $m = 2$ oscillations the current profile should be broadened so that q is only slightly greater than 1 every within the plasma [23]. Stronger modification may increase the current gradient at the $q = 2$ surface causing $m = 2$ oscillations and possible plasma disruption. At the $q = 2$ surface LH waves appear to produce strong $m = 2$ oscillations [23]. Although $m = 2$ oscillations have been eliminated by LH waves near the $q = 3$ surface, for smaller and larger values of q , they are triggered by LH waves [23]. The lack of success in stabilizing $m = 2$ oscillations may be caused by the LH waves being absorbed over a relatively broad region.

To stabilize the ITER plasma against all MHD modes would probably require continuous monitoring, control and adjustment of the LH absorption region during startup. External control of the LH absorption region has been performed in ASDEX. The launched wave spectrum can be gradually adjusted from

$N_{||}$ of 1.9 to 3.2. As $N_{||}$ increases, the current drive efficiency decreases although changes in current profile increase [23]. As $N_{||}$ increases, the RF current drive shifts towards the outside and the central electron heating shifts towards off-axis heating.

6.0 LH EXPERIMENTS ON PETULA-B

LHCD, LHH and sawteeth suppression experiments have been conducted on Petula-B at two RF frequencies, 1.3 GHz and 3.7 GHz. In upgrading the RF frequency from 1.3 GHz to 3.7 GHz, the density cutoff for current drive was increased from $1.5 \times 10^{19} \text{ m}^{-3}$ to $8 \times 10^{19} \text{ m}^{-3}$. In addition, sawteeth oscillations were also stabilized at a higher plasma density (from $3 \times 10^{19} \text{ m}^{-3}$ to $6.5 \times 10^{19} \text{ m}^{-3}$).

6.1 Current-Drive Experiments

Current drive experiments have been performed in the Petula-B tokamak at Grenoble with parameters given in Table 5 [25-27]. LHCD was obtained in plasmas with $\bar{n}_e < 2 \times 10^{19} \text{ m}^{-3}$ with a maximum rampup rate of 100 kA/s (obtained in a $\bar{n}_e = 0.8 \times 10^{19} \text{ m}^{-3}$, $P_{RF} = 200 \text{ kW}$, $I_p = 120 \text{ kA}$) [26]. Current drive efficiencies varied from 0.06 - 0.11 ($\times 10^{20} \text{ A/m}^2/\text{W}$) varying with plasma discharge conditions (gas, limiter, etc.) [26]. Current drive efficiencies, as in other tokamaks such as ASDEX improves when there is some OH driven current. For loop voltage of 0.1 to 0.3 V, the LHCD efficiency improves to .26 ($\times 10^{20} \text{ A/m}^2/\text{W}$). The most efficient current drive has a waveguide phasing of 60°. This maximum agrees with theory assuming that electrons carry the current have energies between 50 - 220 keV (accessibility limit). The current drive efficiency versus magnetic field is shown in Figure (10) from Ref. [26] along with the theoretical curve based on accessibility arguments. Accessibility plays an important role in current drive effectiveness [26].

Current drive studies have been extended to higher densities by using a higher frequency of 3.7 GHz. For low densities ($\bar{n}_e = 0.8 - 1.6 \times 10^{19} \text{ m}^{-3}$) the 3.7 GHz LH power produced the same current drive efficiency as the 1.3 GHz systems [27]. Using LH at 3.7 GHz, current could be driven up to the maximum

TABLE 5

PETULA-B PARAMETERS

	LHCD		LHH	SAWTEETH SUPPRESSION	
R(m)	0.72	0.72	0.72	0.72	0.72
a(m)	0.17	0.17	0.17	0.17	0.17
B(kG)	25-28	28	28	22	22
$\bar{n}_e$ ($\times 10^{19} \text{ m}^{-3}$)	< 1.5	≤ 8	5.8	3	6.5
I_p (kA)	140	100-170	155	170	
q(A)				~ 2.8	2.45
f (GHz)	1.3	3.7	1.3	1.3	3.7
P_{RF} (kW)	120	500	700	120	≥ 240
f_{pulse} (s)	0.25-0.5	0.25-0.5	0.25-0.5	0.25-0.5	0.25-0.5
η_{CD} ($\times 10^{20} \text{ A/m}^2/\text{W}$)	.06-.11	$\sim .1$	---	---	---
	without OH				
	2.6 with residual OH				
η_{HEAT} ($\times 10^{19} \text{ eV/m}^3/\text{kW}$)	---		4.5	---	---
$N_{ }$	2.4	1.5-2.7	4-7		1.7
ΔT_i (eV)			180		

density achievable in Petula-B ($\bar{n}_e \sim 1 \times 10^{20} \text{ m}^{-3}$). However there was not enough available RF power to sustain the plasma current with RF alone for densities above $2 \times 10^{19} \text{ m}^{-3}$. One reason for this is that current drive efficiency appears to scale as $1/n_e$ [27]. Other possible mechanisms involve wave accessibility or absorption saturation. The density limit beyond which current drive is not seen appears to scale as f^a where $1.7 < a < 2$ [27].

6.2 Electron and Ion Heating Experiments

Besides current drive, the LH ion heating regime was examined in Petula-B, while testing a multijunction grill [25]. The ion temperature increased from 450 eV to 630 eV using 250 kW of RF power with $I_p = 155 \text{ kA}$ and $\bar{n}_e = 5.8 \times 10^{19} \text{ m}^{-3}$. An ion heating efficiency of $4.5 \text{ eV} \times 10^{19} / \text{m}^3 / \text{kW}$ was obtained with the multijunction grill, which is comparable to that obtained by conventional grills in Petula-B. During ion heating, the flux of metallic impurities increases until it saturates at 2 - 3 times the initial values. The large increase in radiation causes the electron temperature to remain constant. It seems that fast ions with large orbits are created which hit the walls and limiters. In addition, sputtering could be further enhanced by the LH heating electrons at the plasma edge. Code simulations indicate that 40% of the power is deposited in the bulk ions while 25% is lost due to unconfined fast ions. The energy confinement time appears to be consistent with Kaye-Goldston L-mode scaling [26]. When the plasma density exceeds $6 \times 10^{19} \text{ m}^{-3}$, the power is absorbed by ions in the outer part of the plasma. This does not agree with theory that indicates absorption should occur in the plasma center up to densities of $8 \times 10^{19} \text{ m}^{-3}$. Bulk ion heating is seen over a wide range of $N_{||}$ (from 4 to 7) [26] and the efficiency improves as the plasma current increases.

Electron heating has not been effective in Petula-B [26]. The low energy confinement time for bulk electrons (3 to 5 ms) is smaller than in Alcator C (20 - 40 ms). This short time prevents effective transfer of energy from the fast electron population to the bulk. ECE measurements find a substantial amount of fast electrons in discharges with densities up to $4.8 \times 10^{19} \text{ m}^{-3}$ [26].

6.3 Sawteeth Suppression

Petula-B has also been able to suppress sawteeth using LHCD [26]. Unlike other tokamaks, the energy confinement time is greater in all LHCD discharges than in the corresponding OH ones. This may be caused by the LH waves decreasing the plasma scrape-off density while increasing the density gradients at the edge. For $B = 2.2T$, $I_p = 170$ kA, $q(a) = 2.45$, $\bar{n}_e = 1.25 \times 10^{19} \text{ m}^{-3}$, 120 kW of RF power is required to suppress the oscillations where the $q = 1$ surface lies at $r/a \sim 0.3$ [26].

The increase in the $m = 2$ tearing mode is very sensitive to the LH driven current. The amplitude of this mode is plotted versus $q(a)$ in Figure (11) from Ref. [26]. Although LH triggers $m = 2$ oscillations at $q(a)$ larger than OH discharges, the amplitude at $q(a) = 3$ is reduced thereby allowing the plasma not to disrupt until the RF is turned off.

The 3.7 GHz LH power acts similarly to the 1.3 GHz RF for densities below $3 \times 10^{19} \text{ m}^{-3}$. In moderate density plasmas ($3 - 4.5 \times 10^{19} \text{ m}^{-3}$), the amplitude and period of the sawteeth increase at low RF power [27]. As the RF power is raised, the sawteeth may disappear and then later reappear. For high power levels, they completely disappear. As in Alcator C, the sawteeth inversion radius appears to remain constant for varying amounts of RF power. The fact that the inversion radius remains constant, and that sawteeth may be stabilized even in the presence of $m = 1$ oscillations (as also seen in PLT, implies that LH can stabilize sawteeth, even when q at the center < 1 [27].

At high densities ($\bar{n}_e = 4.5 - 6.5 \times 10^{19} \text{ m}^{-3}$), for RF power levels greater than ~ 240 kW, internal disruptions completely disappear although $m = 1$ oscillations may occur [27]. When $N_{||}$ was changed from 1.7 to 2.7, sawteeth could not be suppressed at the higher densities.

7.0 LH EXPERIMENTS ON THE FT TOKAMAK

LH current drive and heating experiments have been performed on the FT tokamak at Frascati at 2.45 GHz [28] and 8 GHz [29-30]. Using LH with $f = 2.45$ GHz, $T_e(0)$ increased about 1 keV while $T_i(0)$ increased 400 eV using 400 kW of power with plasma parameters given in Table 6 from Ref. [28] wh

TABLE 6

FT PARAMETERS

	TYPE A		TYPE B	
	OH	RF	OH	RF
B_T (kG)	80		60	
I_p (kA)	280	300	310	330
q_a	6.8	6	4.6	4.4
$\bar{n}_e$ ($\times 10^{19} \text{ m}^{-3}$)	3.3	4	3.4	5.2
$T_e(o)$ (keV)	2.0	3.0	1.75	2.4
$T_i(o)$ (keV)	.75	1.2	.95	1.4
Z_{eff}	2.2	2.2	2	2
P_{RF} (kW)		280		280
P_{OH} (kW)	440	330	570	450
P_{RAD} (kW)	150	230	200	280
$(\Delta A/A)_{sx}$	0	0	~25%	~25%
τ (ms) sawtooth	0	0	1.9	3.8

Type A indicates no sawteeth, Type B their presence. The heating efficiency with little energy confinement deterioration defined as $\frac{P_{RF}}{P_{OH}}$ was .6 - .75 for RF power level of 280 kW [28]. For the maximum amount of LH available at 2.45 GHz, ($P_{RF} \leq \frac{3P_{OH}}{4}$) sawteeth were not fully stabilized [29] although their period was increased by a factor of 4 [29]. Since the density limit appears to be proportional to the frequency to some power (~ 2) and FT can operate up to $\sim 5 \times 10^{20} \text{ m}^{-3}$, an 8 GHz LH system was developed [29-30]. For 8 GHz operation a 4 x 4 waveguide coupler was set to have a phasing of 180° or 120° . The plasma parameters were $B = 6 - 8 \text{ T}$, $I_p = 300 - 450 \text{ kA}$, and $\bar{n}_e = 4 - 20 \times 10^{19} \text{ m}^{-3}$. No ion tails were observed though this may be attributed to the low RF power levels ($P_{\text{coupled}} \leq 130 \text{ kW}$). An increase in $T_e(0)$ and $T_i(0)$ of about 100 eV was seen with 90 kW of LH injected into a plasma with $B = 8 \text{ T}$, $I_p = 380 \text{ kA}$, $\bar{n}_e = 1 \times 10^{19} \text{ m}^{-3}$ and phasing = 180° [29].

8.0 LH EXPERIMENTS ON JT-60

More recently LHCD along with neutral beam and ICNF heating have been started on the JT-60 tokamak at JAERI with parameters given in Table 7 from Ref. [31] as achieved by November 1986. The 3 antennas consist of 4x8 waveguides. Experiments have coupled up to 1.4 MW at $f = 2.0 \text{ GHz}$ with $I_p = 0.5 - 2.0 \text{ MA}$ with $\bar{n}_e = (0.2 - 4 \times 10^{19} \text{ m}^{-3})$, $B = 40 \text{ kG}$ and $q_a = 3.7 - 11$ [33]. Coupling efficiencies from 0.75 - 0.85 have been achieved, depending upon the distance from the coupler to the plasma [34]. The reflection coefficient remains constant as the power is increased from 0.5 MW to 1.5 MW.

8.1 Current-Drive Experiments

With an antenna phasing of 90° and $N_{||} \sim 1.7$ [31] a current drive efficiency $\eta_{CD} = \frac{n_e I_{RF}}{P_{RF}}$ was .1 - .17 ($\times 10^{20} \text{ A/m}^2/\text{W}$) without neutral beams was obtained and increased to (.2 - .28) ($\times 10^{20} \text{ A/m}^2/\text{W}$) with beams as shown in Figure (12) from Ref. [31]. With 1 MW of LH power, approximately 1 MA of current is driven in a plasma with $\bar{n}_e \sim 0.37 \times 10^{19} \text{ m}^{-3}$ and $B = 40 \text{ kG}$ [32].

TABLE 7

DESIGNED AND ACHIEVED PARAMETERS OF JT-60(a) JT-60 Parameters

	Designed	Achieved	
		(Divertor)	(Limiter)
Major Radius	3.0 m	3.10 m	3.04 m
Minor Radius	0.95 m	0.90 m	0.90 m
Toroidal Field*	4.5 T	4.5 T	4.5 T
Plasma Current	2.7 MA**	2.0 MA	2.0 MA
Pulse Length	10 s	10 s	10 s

* At a major radius of 3 m

** 2.1 MA with divertor

(b) JT-60 NBI Parameters

	Designed	Achieved	
		(Test)**	(With plasma)
Total power into torus	20 MW	20 MW	20 MW
Number of beamlines	14	14	14
Number of sources	28	28	28
Beam energy	75 keV	~ 100 keV	>75 keV
Species power ratio*	--	80:13:7	75:16:9
Pulse length	10 s	10 sec	2 s

* $H^0(E):H^0(E/2):H^0(E/3)$ at 75 keV

** Each beamline tested separately in a test stand

(c) JT-60 RF Parameters

	Designed	Achieved	
		(Test)*	(With plasma)
<u>Lower Hybrid</u>			
Frequency	1.7-2.23 GHz	1.7-2.23 GHz	2.0 GHz
Number of units	3	3	1
Power tubes per unit	8 klystrons	8	8
Power at sources	24 MW	24 MW	2.0 MW
Power into torus	15 MW	--	--
Power into plasma	7.5 MW	--	--
Pulse length	10 s	10 s	0.75 s
<u>Ion Cyclotron</u>			
Frequency	110-130 MHz	110-130 MHz	120 MHz
Number of units	1	1	1
Power tubes	8 Tetrodes	8	8
Power at source	6 MW	6 MW	1.6 MW
Power into torus	5 MW	--	1.4 MW
Power into plasma	2.5 MW	--	--
Pulse length	10 s	10 s	1.2 s

* Each unit tested separately

The maximum amount of current sustained by the LH has been 1.7 MA using 1.2 MW of RF power at $f = 2.0$ GHz [32-33] with a total plasma current of 2.0 MA. As seen in other tokamaks (i.e., ASDEX) the current drive efficiency scale as $1/n_e$ as shown in Figure (13) from Ref. [32].

Unlike ASDEX, when neutral beams are added the current drive efficiency nearly doubles [32]. Because of the necessity in invoking a filling mechanism to close the gap between the electrons which interact with the launched LH waves and those needed to supply the current, it is postulated that higher electron temperatures would narrow this gap, resulting in higher current drive efficiencies [32-33]. Figure (14) from Ref. [33] shows a typical OH + LH + NBI discharge where 0.88 MW of RF power is launched into a plasma with $I_p = 1$ MA. Before the 8.8 MW of beams are injected, the LH drives 1.25 MA. After the beams are on, the RF driven current drops to 0.75 MA while the electron density increases from $0.35 \times 10^{19} \text{ m}^{-3}$ to $0.9 \times 10^{19} \text{ m}^{-3}$ [33]. This indicates that the LHCD efficiency increases from 1.5 to 2.4 [33].

In JT-60 with $P_{RF} = 0.57$ MW $I_p = 0.7$ MA and $\bar{n}_e \sim 0.2 \times 10^{19} \text{ m}^{-3}$, the OH coil was recharged at a rate of 1.3 kA/s giving a recharging efficiency $\epsilon = \frac{I_p I_f M}{P_{RF}} = 0.2$ where I_f is the rising time of the OH coil current and M is the mutual inductance between the OH coil and the plasma equal to $140 \mu\text{H}$ [33]. Figure (15) from Ref. [33] shows the current rampup efficiency obtained of 0.2 in JT-60 where

$$\eta_{\text{RAMP}} = \frac{\frac{dW_{RF}}{dt} - P_{\text{EXT}} + D}{P_{RF}} \quad (7)$$

and

$$\frac{dW_{RF}}{dt} = \frac{d}{dt} \left(\frac{1}{2} L_p I_p^2 \right) \quad (8)$$

and D is a dissipation term [6,33]

$$D = I_d V_c + V_c^2/R \quad (9)$$

where I_d is the driven current, V_c its dissipation normalized to an effective

one turn loop voltage, V the axial loop voltage of the electric fields and R the effective plasma resistance including the superthermal current carrying electrons [6]. PLT had obtained a similar rampup efficiency ($\eta_{\text{RAMP}} = 0.1 - 0.2$) [6,33].

When one LHCD antenna and two LHH couplers ($N_{||} = 1.3$ $\Delta\phi = 120^\circ$) are used 2 MA of plasma current can be driven with $\bar{n}_e = 0.32 \times 10^{19} \text{ m}^{-3}$ and $B = 45 \text{ kG}$ for 2.5 s [34-36]. A total injected power of 3.1 MW at 2.0 GHz was used with 0.8 MW being launched from the LHCD antenna. With the increased LH power, the current drive efficiency without beams ranges from 0.9 to 2.0 with $T_e(0)$ 2 - 3 keV while with beams the range is from 1.5 to 3.0 with $T_e(0)$ increasing to 3 - 4 keV [34].

8.2 LH Heating Experiments

In JT-60 for LH at 2 GHz, the density limit is predicted to be at $\bar{n}_e = 4 \times 10^{19} \text{ m}^{-3}$ in hydrogen and $8 \times 10^{19} \text{ m}^{-3}$ in helium [34]. Experiments were performed in helium plasmas with the two LHH couplers and the LHCD coupler at densities ranging from 1.0 to $4.0 \times 10^{19} \text{ m}^{-3}$. A maximum heating efficiency of $\eta_{\text{HEAT}} = 2 \times 10^{-19} \text{ eV/m}^3/\text{kW}$ was obtained with a maximum electron temperature of $T_e(0) = 6 \text{ keV}$ achieved at $n_e = 1.7 \times 10^{19} \text{ m}^{-3}$ with $P_{\text{LH}} = 2.4 \text{ MW}$. An increase in the ion temperature up to 1 keV was observed with no hot ion tails present.

8.3 Confinement, Current Profile and MHD Stabilization Studies

As in smaller tokamaks, $m = 1$ oscillations are stabilized in JT-60 [32]. When neutral beams are injected into a LH current driven plasma, the energy confinement time increases 20 - 30%. When 11 MW of neutral beam power is injected, the confinement time approaches that obtained by OH discharges and is 2.5 times greater than Kaye-Goldston L-mode scaling [32]. The particle confinement time is also increased.

JT-60 finds that low powers of RF tends to sharpen the current profile while high power levels ($P_{\text{RF}} > 2n_{19}I_p$) tends to flatten it [33]. This further suggests that LH could be used to control the current profile in ITER. Sawtooth oscillations were suppressed for $P_{\text{RF}} = 450 \text{ kW}$ injected into a 1 MA plasma.

9.0 LH EXPERIMENTS ON OTHER TOKAMAKS

In this section we note some results from LH experiments on other tokamaks. Current has been driven, the particle confinement improved and the density limit varying with frequency and magnetic field has been examined in Versator II [19]. The temperature dependence of the current drive efficiency has been investigated in the T-7 tokamak at Kurchatov [37] with up to 900 kW of power with $f = 900$ MHz into a plasma with $R = 1.22$ m, $a = .35$ m, $B = 23$ kG and $I_p \leq 200$ kA. The antenna was phased at 90° with $N_{||} \sim 3$. The current drive efficiency, including temperature effects may be expressed as [37]

$$\eta_{CD} = \frac{Z_{eff} T^5}{6} \frac{\bar{n}_e I_{LH} R}{P + I_{LH} V_L} = \left(\frac{m}{18\pi^2 e^3 \ln \Omega} \right) \left(\frac{v_2^2 - v_1^2}{\ln(v_2/v_1)} \right) \quad (10)$$

where V_L is the loop voltage, I_{LH} is the driven current and v_1 and v_2 are the limits of the quasi-linear plateau on the electron distribution function. Results from these experiments along with Equation (10) plotted with a dashed line is shown in Fig. (15) from Ref. [37]. Besides current drive electron heating ($\Delta T_e(0) = 1$ keV) was observed with n_e rising 30 - 40% for $P_{RF} = 160$ kW and $I_p = 170$ kA.

In addition to LHCD, LH and ECRH have been combined successfully in T-7 [37], JFT-2M [38], WT-2 [39-40] and WT-3 [39] and JIPP T-IIU [41] tokamaks.

10.0 EXTRAPOLATION TO ITER AND SUMMARY

Attempts to heat ions with LH waves have been met with mixed results. Significant ion heating has been seen in JFT-2, WEGA, JIPP-T-II and Petula-B [26]. Table 8 is a summary of the LH antennas used in these experiments [43]. PLT and Versator have seen some ion heating while Alcator C and FT have not seen bulk ion heating [26]. Petula-B has produced 0.45 keV ion temperature rise using 700 kW of power [26]. For ITER, with proposed parameters given in Table 9, with its high density and temperature would not be operated in the ion heating regime. Any ion heating would be near the plasma edge.

TABLE 8

SUMMARY OF LOWER HYBRID GRILL COUPLERS ON TOKAMAKS

Machine	Freq. (GHz)	No. of Wgds per grill	Wgd dim. (cm)	Wgd Material	Coating	$\omega_{ce} > \omega$?	Additional Pumping?	Max. power density (kW/cm ²)
JFT-2	.65, .75	4	3.3 x 29	SS	Cu, Ti	No	Yes	1.8
ATC	.8	4	2.5 x 20	SS	--	Yes	No	0.6
PLT	.8	6	3.5 x 22	SS	C	Yes	No	1.6
Versator II	.8	4	2.5 x 24	SS	--	No	No	1.0
Versator II	.8	4	1 x 21	SS	C	No	No	1.6
JIPP-T II	.8	2	14 x 68 (c-shaped)	SS		Yes	No	1
Wega	.8	4	2 x 24.4	Ti	Ti	No	--	2
				SS		No		
T-7	.9	3	2 x 22.0	SS	Ti	No	No	1.9
WT-II	.915	4	1.3 x 18.5		--		No	0.5
Petula B	1.25	4	1.8 x 16.5	Ti	Ti	No	Yes	3.4
Petula B	1.25	4	1.9 x 16.5	SS	--	No	No	3.4
Asdex	1.3	8	3.0 x 16.5	SS	Cu, Au	No	Yes	5.4
JT-60	1.7-2.3	32 (8 x 4)	3 x 11.5, 1.8 x 11.5	SS	Cu	No	Yes	4.5 (planned)
Alcator A	2.45	2	1.3 x 8.1	SS	--	Yes	No	4.5
Alcator A	2.45	2	1.3 x 8.1	SS	--	No	No	8
FT	2.45	4 (2 x 2)	1.5 x 7.1	SS	--	Yes	No	6
PLT	2.45	8	1.0 x 10.0	SS	--	Yes	No	5 (planned)
Versator II	2.45	4	1.0 x 8.6	SS	--	No	No	2.5
Alcator C	4.6	16 (4 x 4)	0.8 x 5.8	SS	--	Yes	No	9

TABLE 9

U.S. ITER NOMINAL BASELINE

(December 1987)

R (m)	4.04
a (m)	1.41
B (kG)	49.8
I _p (MA)	15.5
P _{FUS} (MW)	630
q (a)	~ 3.5
P _{NB} (MW)	88
$\bar{n}_e$ ($\times 10^{19} \text{ m}^{-3}$)	8
T _e (keV)	21

RF PARAMETERS

$\bar{n}_e$ ($\times 10^{19} \text{ m}^{-3}$)	3.5
(average density from $\frac{r}{a} = 1$ to $\frac{r}{a} = .58$)	
T _e (keV)	9
(density-weighted average temperature from $\frac{r}{a} = 1$ to $\frac{r}{a} = .46$)	
η_{CD} ($\times 10^{20} \text{ m}^2/\text{W}$)	1.2 - 1.5
I _{RF} (MA)	4.5 - 5.6
f (GHz)	3.7

$$1.2 \lesssim N_{||} \lesssim 2$$

If the results from JT-60 [32] can be extrapolated to ITER, then LHCD can greatly enhance the plasma performance. The current drive efficiency was nearly doubled in the presence of neutral beams, $m = 1$ oscillations are stabilized and the decreased energy confinement time seen in neutral beam heated plasmas is increased in LH driven current discharges in the large tokamak JT-60. As seen in JT-60 and T-7, the current drive efficiency appears to increase with electron temperature, so that one might expect the current drive efficiency in ITER to be greater than in present day experiments. Figure (16) from Ref. [33] plots the temperature dependence of the current drive efficiencies obtained in various LH driven tokamaks. The electron temperature in JT-60 in OH + LHCD discharges is $T_e \approx 1.05 - 2.5$ keV while this increases to $1.45 \sim 3.1$ keV when neutral beams are added. The dashed lines correspond to the theoretical calculation of current drive efficiency $\hat{j}/P_d$ where

$$\frac{\hat{j}}{P_d} = \frac{\tau}{\tau_0 \left(\frac{v_{te}}{w/k} \right)} \quad (11)$$

and

$$\tau_0 = \frac{4\pi \epsilon_0^2 m_e^2 v_{te}^3}{n_e e^4 \ln \Lambda} \quad (12)$$

$$\tau = \frac{6\pi(2\pi)^{1/2} \epsilon_0^2 m_e^{1/2}}{T_e^{3/2} n_i z_i^2 e^4 \ln \Lambda} \quad (13)$$

showing an electron temperature dependence on the current drive efficiency.

LH has stabilized sawteeth in several tokamaks using current drive phasing. In the higher density tokamaks (Alcator C and Petula-B) the sawtooth inversion radius does not vary with RF power. PLT and Petula-B observed $m = 1$ oscillations even when sawteeth were stabilized implying that q may be below 1 in the plasma. This may indicate that these oscillations may be stabilized by current profile modification outside of the $q = 1$ surface. This has an important impact on ITER because the large size, high temperatures and

densities will not allow the LH waves to reach the center (hence the predicted $q = 1$ surface). ASDEX has demonstrated that the launched spectrum can be gradually adjusted. As $N_{||}$ increases, the heating shifts towards the outside while the current profile changes shape. This indicates that LH can be adjusted during an ITER run with some feedback mechanism so that sawteeth and perhaps $m = 1$ and $m = 2$ oscillations are eliminated.

Tokamaks such as PLT and Petula-B have successfully used LH waves to rampup the plasma current. Petula-B has ramped the current up at a rate of 100 kA/s while PLT initially ramps up at ~ 200 kA/s and then falls to 35 kA/s. In addition, JT-60 has demonstrated that the OH coil can be recharged (at a rate of 1.3 kA/s). Therefore, LHCD can be used to assist ITER when operated in the pulsed mode by saving OH volts-seconds during startup and lessening cyclic fatigue effects in the recharge phase.

In addition, as seen in JT-60, the use of neutral beams increase the LHCD efficiency. As envisioned in ITER, neutral beams will drive current in the plasma center while LH will drive current in the outer region as well as control the current profile to eliminate MHD oscillations and assist in disruption control. JT-60 has observed that the neutral beams themselves may eliminate sawteeth. This may occur in ITER as well since the neutral beam power will be large. In addition LH has eliminated the increased sawteeth due to central ICRF ion heating. The ICRF ion heating is predicted to simulate the effect of hot alphas being created at the plasma interior and quickly slowing down. Therefore, it is hoped that LH can prevent sawtooth oscillations from occurring in the burning ITER plasma.

On the other hand, it has been observed that the current drive efficiency decreases as n_e increases (as $1/n_e$). The higher steady-state electron density expected in ITER will tend to decrease the current drive efficiency. The amount of LH power required in present day experiments to stabilize sawteeth has ranged from 20% of the required RF power needed to drive the entire plasma current in Alcator C to 66% in ASDEX. In ITER, these fractions indicate that a large amount of RF power is required.

LH waves also often trigger $m = 2$ oscillations when OH discharges do not as observed in ASDEX. The relatively broad region of absorption in present day experiments make the elimination of $m = 2$ oscillations difficult. The use of LH to eliminate sawteeth and $m = 1$ oscillations in ITER must be carefully

controlled so that $m = 2$ oscillations are not triggered (as observed in ASDEX).

Also ASDEX has seen that the LH power has been absorbed by the neutral beam ions, thereby reducing the overall LHCD efficiency. Since these results contradict those obtained in JT-60, it is important that this be resolved before ITER is designed.

Finally, there is a density limit above which LHCD is not observed as shown in Table 10 and Figure 17 from Ref. [42]. It is not completely understood at this time the cause of this effect. Ref. [42] suggests that in the course of propagating through the plasma, the LH $N_{||}$ spectrum is upshifted until at high enough density, mode conversion to a hot plasma mode occurs. This wave then propagates back towards the plasma edge and damped by perpendicular processes. No parallel electron tails are produced so no current drive results. The density limit predicted by this process is shown as a dashed line in Fig. 17 and listed in Table 10. Although the present day data base can be extrapolated to ITER parameters, the theoretical investigation of density limits and current drive efficiencies needs further work.

TABLE 10

DENSITY LIMITS OBSERVED IN VARIOUS TOKAMAK LOWER HYBRID EXPERIMENTS

	f	B	T _i	T _e	n _{observed}
	Mhz	kG	eV	eV	10 ¹⁸ m ⁻³
JFT-2 (D gas)	750	14		250	6
Versator	800	10	120	350	7
Versator	2450	10	120	350	10
PLT (D gas)	800	30			10
WT-2	915	11	50	200	7
Aicator-C	4600	100			100
FT (D gas)	2450	40			45
FT (D gas)	2450	60			75
FT (D gas)	2450	80			60
WEGA	800	22.5			12
WEGA (D gas)	800	22.5			18
T7 (D gas)	900	19			17
Petula-B	1300	28		1000	18
Petula-B	3700	28		1000	80
JIPP-TII	750	14			8
ASDEX	1300	22			20
ASDEX (D gas)	1300	22			30
JT-60	2000	45			20
Encore (Ar gas)	450	1.5	5	10	1

APPENDIX 1

INTERMEDIATE LOWER HYBRID OPERATION FOR ITER

L. John Perkins and Joel H. Schultz

1. MOTIVATION

Pure inductive operation, an essential requirement for ITER, is based upon a good experimental and theoretical foundation but is presently understood to give rise to operational issues which leave much to be desired: e.g., burn stability control, sawteething, disruption initiation, non-optimum profiles for equilibrium and beta, pulsed losses and fatigue, short burn times, and low end-of-life fluences; the last factor is of particular importance. By contrast, non-inductive, steady-state current-drive offers an attractive reactor operating scenario but there are several potential problems which could undermine its viability, e.g., low efficiency, low Q operation, high injected power, the possible non-existence of bootstrap current, driven plasma instabilities, high divertor heat loads under low density conditions, and hardware development of 1-3 MeV beam systems at power levels of many tens of MW.

Accordingly, the key question to be addressed here is:

Can the use of lower hybrid (LH) current-drive in conjunction with inductive drive result in an attractive, contingency operating scenario for ITER in terms of:

- (i) Startup assist: reduced pulsed losses; volt-seconds savings for burn;
- (ii) Burn-assist: high Q operation; current-profile control; burn-stability; sawteeth and disruption suppression; long burn times with fully established profiles; high fluence;
- (iii) Current maintenance during transformer reset: reduced stress amplitudes, pulsed losses and fatigue cycling?

In this way we are able to combine the advantages of the tried and tested, highly efficient inductive drive (factor of ~ 100 greater than RF or NB methods) with a modest amount (several MWs) of LH power with its good experimental database to considerably enhance the flexibility of ITER operating scenarios.

In addition, although the current ITER Terms-Of-Reference call for a machine capable of full transformer capability with up to 20 MA of inductively-driven current, we will also determine whether the utilization of inductive + LH operation would result in a smaller, cheaper machine with lower total volt-seconds requirement.

We should stress that, given the undesirable operational features of a pure inductive, ignited machine and the present uncertain evolution of pure steady-state current-drive methods, it is important that we adequately assess this important contingency mode. Given the current uncertainty in the physics and technology database for all operating scenarios, operational flexibility of ITER is probably one of its most important attributes at this stage in the design process.

2. POTENTIAL IMPACT FOR ITER

The major potential impact is the addition of three intermediate modes of operation (#2, 3 and 4 -- see Table 1) in between pure inductive operation and pure steady-state operation which would do much to enhance the overall flexibility of ITER. Note, from Table 1, that the transition from basic inductive operation through full steady-state operation, brings increasing benefits in terms of longer burn times, higher fluences and reactor attractiveness but at the expense of greater uncertainty in the physics and technological database. Accordingly, the Intermediate LH scenarios provide a contingency against the detrimental operating conditions perceived for inductive ignition, and the potential problems and uncertainties currently associated with full current-driven operation.

IMPACT OF THE INTERMEDIATE LH OPERATING SCENARIOS

<u>OPERATING MODE</u>	<u>PERCEIVED DISADVANTAGES</u>	<u>PERCEIVED ADVANTAGES</u>
1. PURE INDUCTIVE OPERATION; IGNITED BURN	Pulsed losses; sawteeth and disruptions; time-dependent current profile evolution; short burn time; pulsed fatigue; low cycle limit; low fluence; burn control stability.	Good experimental database (for inductive operation); high efficiency; simple operation; least (present-day) uncertainty.
2. LH RAMPUP; IGNITED INDUCTIVE BURN	Sawteeth and disruptions; time-dependent profile evolution; short burn time; low fluence; burn control stability.	Slow current ramp; reduction of pulsed losses and fatigue; increased volt- seconds for burn.
3. LH RAMPUP; LH BURN-ASSIST	High Q only (i.e., not $Q = \infty$); LH system requirement for long pulse operation.	Current-profile control; sawteeth and disruption suppression; reduced impact of confinement uncertainties; high Q stable operation; long burn times; increased fluence.
4. LH BURN-ASSIST; CURRENT MAINTENANCE DURING TRANSFORMER RESET	As 3 with addition of: possibility of reduced burn times due to high-low beta transitions; long transformer reset times.	As 3 with addition of: reduced cyclic stress fatigue and pulsed losses; increased cycle limit; high fluence.
5. PURE NON-INDUCTIVE, STEADY- STATE CURRENT DRIVE	Moderate-low Q operation; high injected powers; uncertainty of bootstrap current; hardware development.	Full steady-state operation; negation of pulsed fatigue limits; maximum fluence capability; significant reactor relevance.

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Figure 1 Time evolution of plasma current, plasma density, vertical field, and poloidal beta plus internal inductance using 800 MHz in PLT.

Figure 2 Current ramp up of various power levels in PLT with $\bar{n}_e = 2.2 \times 10^{18} \text{ m}^{-3}$ and for 800 MHz and $N_{||} = 1.5$.

Figure 3 Current drive efficiency $\frac{I_{13} \bar{n}_e R}{P_{RF}}$ versus $\bar{n}_e$ for 800 MHz 6-element wide and narrow grills and 2.45 GHz 8-element and 16-element grills in PLT.

Figure 4 T_e profile in PLT measured by Thomson scattering at 600 ms compared with OH alone using 500-600 kW of LH at various waveguide phasings. Phasing of -90° , -105° , -120° , and -180° corresponds to $N_{||0}$ of 1.8, 2.1, 2.4, and 3.6 respectively.

Figure 5 Amplitude of $m = 0$ (sawteeth) of $r = 0$ and $m = 1$ at $r = -5$ cm and 6 cm vs RF power is varied as determined from second electron cyclotron harmonic emission in PLT with $\bar{n}_e \sim 10^{19} \text{ m}^{-3}$, $B = 29$ kG, $I_p = 500$ kA and $f = 2.45$ GHz.

Figure 6a Global confinement times for 130-160 kA LHCD discharges (solid circles) and OH discharges (open circles) versus density.

Figure 6b Global confinement times for LHCD discharges (solid circles) and OH discharges (open circles) versus toroidal magnetic field.

Figure 7 Increase in $T_e(o)$ and $T_i(o)$ for LHCD ($\Delta\phi = 90^\circ$) and anti-current drive ($\Delta\phi = -90^\circ$) versus P_{RF} in Alcator C with $\bar{n}_e = 0.7 - 0.9 \times 10^{20} \text{ m}^{-3}$ and $q(a) = 4.7$. Solid points indicate stabilized sawtooth discharges.

Figure 8 Average sawtooth period versus P_{RF} for LHCD anti-current drive and symmetric ($\Delta\phi = 180^\circ$) phasings in Alcator C with $\bar{n}_e = 0.7 - 0.9 \times 10^{20} \text{ m}^{-3}$ and $q(a) = 4.7$.

NI + LHCD

Figure 9 Decrease in OH power input with LHCD in combination with neutral beams versus neutral beam power in ASDEX.

Figure 10 Current drive efficiency versus $(\frac{\omega_{pe}}{\omega_{ce}})^2$ in Petula-B.

Figure 11 Amplitude of $m = 2$ tearing mode versus $q(a)$ with LHCD (solid circles) and without (open boxes) in Petula-B.

Figure 12 I_{RF} versus $P_{LH}/\bar{n}_e$ for OH + LHCD (solid circles) and OH + LHCD + NBI (open circles) in JT-60.

Figure 13 $1/\bar{n}_e$ variation of current drive efficiency seen in JT-60

$$I_{RF} = I_p \frac{|\Delta V_e|}{v_e}.$$

Figure 14 Typical JT-60 discharge with OH + LHCD + NBI.

Figure 15 $\dot{W}_{PF} - P_{ext}$ versus P_{RF} in the JT-60 rampup experiment. Slope of line indicates $\eta_{ramp} = 0.2$. Plasma parameters for open circle E2815 were $I_p = 0.5$ MA, $\bar{n}_e = 0.28 \times 10^{19} \text{ m}^{-3}$ while for the solid circle E2809 they were $I_p = 0.7$ MA, $\bar{n}_e = 0.22 \times 10^{19} \text{ m}^{-3}$. The point ■ is that value when evolved with the same plasma conditions of E2815.

Figure 16 LHCD efficiency versus T_e in T-7. Formula (10) is shown by dashed line.

Figure 17 Temperature dependence of $\eta_{CD} = \frac{\bar{n}_e R I_{RF}}{P_{LH}}$. Open circle corresponds to $\bar{T}_e$ before RF injection, open triangle after. Solid figures are estimated temperatures. Dashed lines are calculated current drive efficiencies (Formula 11-12).

Figure 18 Density limits observed in various experiments. Solid circles indicate hydrogen, open circles deuterium and triangles, Argon. Solid line is plot of Eq. (5) from Ref. [42] for hydrogen and dashed line for deuterium. Calculated values are given in Table 9. The dotted line is a plot of a driving model by Wegrowe and Engelmann for hydrogen with $T_i/T_e = 1/3$.

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LOWER HYBRID SLOW WAVE CURRENT DRIVE FOR
STEADY STATE OPERATION OF ITER

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ITER Physics Preparatory Meeting
Fusion Engineering Design Center, Oak Ridge TN
March 10-11, 1988.

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LOWER HYBRID CURRENT DRIVE ISSUES

O Partial penetration into the plasma

- Limits total lower hybrid (LH) driven current.
- Constrains contribution of LH current to the total current density profile.

O Damping on fast alpha particles

- Reduces the effective power damped on electrons (current drive) per unit power launched.

O Narrow $N_{||}$ spectrum requirement

- Large antenna with a very large number of waveguide slots is required.

PLOTS FOR LH CURRENT DRIVE

15

LOWER HYBRID CURRENT DRIVE CALCULATIONS

O Bonoli-Englade ray-tracing and Fokker-Planck code used.

- Shifted circular model equilibrium flux surfaces.
- Fixed density, $n_e(r)$, and temperature, $T_e(r)$, radial profiles.
- Full relativistic, quasi-linear wave damping.
- Damping on a model fast alpha distribution included.

O Bulk plasma parameters determined by the MUMAK power balance code.

RECOMMENDED LH CURRENT DRIVE PARAMETERS

O Frequency: ≥ 8 GHz

- Avoids damping on fast alphas.
- Avoids parametric instabilities at plasma edge.
- Data will be available on 8 GHz operation of a LH system from the FTU experiment this year.

O Wave Spectrum: as narrow as possible

- Maximum penetration into the plasma.
- Tailoring of the current profile by choice of $N_{||0}$.

O Launcher location: outboard midplane

- Assuming double-null divertor configuration.
- For single-null configurations, bottom launch may improve penetration.

I. ANALYSIS :

A. Choice of f_0 to avoid alpha-particle damping :

$$\frac{1}{2} n_a v_{th}^2 > E_0 \quad (1)$$

$$v_{th}^2 \approx (\omega / k_\perp)^2 \quad (2)$$

$$k_\perp^2 \approx \frac{\omega_{pe}^2}{\omega^2} k_\parallel^2 \quad (3)$$

Combining (1), (2), and (3) yields :

$$f_0 \gtrsim f_{pe} n_{max} \left(\frac{m_e}{m_i} \right)^{1/2} \left(\frac{E_0}{\frac{1}{2} m_e c^2} \right)^{1/2} \quad (4)$$

where : $n_i \approx n_{max} = \epsilon_z^{1/2} + \omega_{pe} / \omega_{ci} \quad (5a)$

$$\epsilon_z = 1 + (\omega_{pe} / \omega_{ci})^2 - (\omega_{pi} / \omega)^2 \quad (5b)$$

and : n_{max} is a weak function of $f_0 = \omega_0 / 2\pi$ for $(\omega_{pi} / \omega)^2 \ll 1$

For Typical ITER Parameters :

$$n_e(0) = 1.754 \times 10^{20} \text{ m}^{-3}$$

$$B = 5T$$

(50/50) D-T Mixture

$$n_{max} \approx 2.14$$

WE LAUNCH

$$N_{ii} \approx 1.70$$

FOR OPTIMIZE CASES

$$f_0 \gtrsim 11 \text{ GHz} \quad (\text{using Eq. 4})$$

$$\Rightarrow f_0 \gtrsim 9 \text{ GHz}$$

P. BONOLI
Q-Q6 EXCHANGE
VS/JAPAN
ADVANCED
CURRENT DRIVE

PARTIALLY OPTIMIZED CASE FOR US ITER

O Power spectrum: $P(N_{||}) \propto \exp [-96 (N_{||} - N_{||0})^2]$, $N_{||0} = 1.7$

O Frequency: $f_0 = 8$ GHz, Outside launch

O Penetration: $r_{\max}/a = 0.7$ $r_{\text{peak}}/a = 0.8$

O Profiles: $K(r) = K(0) [1 - (r/a)^2]^{\alpha_x}$, $x = n, T$

$\alpha_n = 1.0$, $\alpha_T = 0.5$

$n_e(0) = 1.6 \times 10^{20} \text{ m}^{-3}$, $T_e(0) = 25.7 \text{ keV}$

O Figure-of-merit: $\eta_{lh} = n(r_{\max}) I_{lh} R_0 / P_{lh} = 0.46$

RAY TRACING RESULTS - $N_H = 1.7$

RAY TRAJECTORY-POLOIDAL PLANE

POLOIDAL PROJECTION

FOKKER PLANCK RESULTS

- POWER DENSITY

CURRENT DENSITY

ELECTRON DISTRIBUTION

$I_{LH} = 2.6 \text{ MA}$
 $P_{LH} = 20 \text{ MW}$
 $\eta = \eta(n_{max}) I_{LH} R_0 / P_{LH} = 0.46$

$\eta_c(n_{max}) = 0.87 \times 10^{-3}$
 $T_e(n_{max}) = 19.0 \text{ keV}$

NARROW WAVE SPECTRUM IS REQUIRED

— MAXIMIZES PENETRATION

— ALLOWS J_{int} TO BE OPTIMIZED

$$\eta_{int} \approx \eta_c(n_{max}) I_{cw} R_c / P_{cw}$$

$I_{cw} = 3.6 \text{ MA}$, $P_{cw} = 20 \text{ MW}$, $\eta_{cw} = 0.46$

$I_{cw} = 3.8 \text{ MA}$, $P_{cw} = 20 \text{ MW}$, $\eta_{cw} = 0.48$

Wavelength (cm) $\eta_c(n_{max}) = 0.57 \times 10^{-3}$

Wavelength (cm)

$\eta_c(n_{max}) = 0.63 \times 10^{-3}$

VARIATION OF LH CURRENT DENSITY PROFILE WITH POWER SPECTRUM CENTRAL N_{110}

GEDANKEN EXPERIMENT: $T_{e0} = T_{i0} = 15 \text{ keV}$; $B_0 = 12 \text{ T}$; $f_0 = 8 \text{ GHz}$

$$\Delta R_H = 0.117$$

P. BONOLI

Q-96 EXCHANGE

US/JAPAN

ADVANCED

CURRENT DRIVE

GEDANKEN EXPERIMENT: $T_{e0} = T_{i0} = 15 \text{ keV}$; $B_0 = 12 \text{ T}$; $f_0 = 8 \text{ GHz}$

$$\Delta R_H = 0.117$$

$$n_e(n_{max}) = 1.6 \times 10^{20} \text{ m}^{-3}$$

$$\eta_{LH} = \eta_e(n_{max}) I_{LH} R_0 / P_{LH}$$

$$\eta_{LH} = 0.50$$

SECOND GENERATION U.S. ITER STEADY STATE PARAMETERS

$Q = 9.2$ $P_{\text{TOTAL}} = P_{\text{NB}} + P_{\text{LH}} = 93 \text{ MW}$ $\langle T \rangle = 1.7 \text{ MW/m}^2$
 NOTE: Neutral beam induced Alfvén instabilities are NOT included.

$R_o = 4.0 \text{ m}$	$I_p = 15.5 \text{ MA}$	$N_{ } = 1.7$
$a = 1.41 \text{ m}$	$I_{bs} = 5.2 \text{ MA}$	$f_o = 8.0 \text{ GHz}$
$\kappa_d = 2.3$	$I_{nb} = 7.8 \text{ MA}$	$\Delta_H = 0.2$
$\delta_d = 0.6$	$P_{nb} = 70 \text{ MW}$	$\eta_{lh} = 0.41 \text{ A/Wm}^2 *$
$B_o = 5.0 \text{ T}$	<u>$E_{nb} = 1.3 \text{ MeV}$</u>	$\eta_{nb} = 0.46 \text{ A/Wm}^2 *$
$\beta_t = 6.6 \%$	$P_f = 855 \text{ MW}$	$\eta_{cd} = 0.68 \text{ A/Wm}^2 *$
$\beta_p = 1.07$	$\langle T \rangle = 1.7 \text{ MW/m}^2$	$I_{lh} = 2.5 \text{ MA}$
<u>$H_T = 1.2$</u>	$\tau_E = 1.1 \text{ sec.}$	$P_{lh} = 23 \text{ MW}$

$$* \eta_{LH} = n_e(n_{max}) I_{LH} R_o / P_{LH}$$

$$* \eta_{CD} = \langle n_e \rangle I_p R_o / (P_{NB} + P_{LH})$$

$$* \eta_{NB} = \langle n_e \rangle I_{NB} R_o / P_{NB}$$

RESOLUTION OF REMAINING ISSUES

0 Issue: Ray tracing and penetration in full 2-D equilibrium geometry

Resolution: Combine generalized LH ray tracing with SELENE (JAERI) 2-D equilibrium solution.

LH RAY TRACING - P.T. BONOLI (MIT)

EQUILIBRIUM FITTING - G.A. SMITH (LLNL)
A.S. DEVOTO (LLNL)

INTERFACE - M.E. FENSTERMACHER (TAN/LLNL)

0 Issue: Consistency between total NBI, LH and bootstrap driven current profiles and the equilibrium flux surfaces.

Resolution: ACCOME (JAERI) code for NBI current drive plus SSRFCVM (MIT) code for LH current drive will be iterated with SELENE (JAERI) until converged.

ACCOME - A.S. DEVOTO (LLNL)

PARAMETRIC VARIATIONS - M.E. FENSTERMACHER (TAN/LLNL)

E Advanced Blanket Modeling

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ADVANCED FUSION MHD POWER CONVERSION

USING THE CFAR CYCLE CONCEPT

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ABSTRACT

The CFAR (compact fusion advanced Rankine) cycle concept for a tokamak reactor involves the use of a high-temperature Rankine cycle in combination with microwave superheaters and nonequilibrium MHD disk generators to obtain a compact, low-capital-cost power conversion system which fits almost entirely within the reactor vault. The significant savings in the balance-of-plant costs are expected to result in much lower costs of electricity than previous concepts. This paper describes the unique features of the CFAR cycle and a high-temperature blanket designed to take advantage of it as well as the predicted performance of the MHD disk generators using mercury seeded with cesium.

INTRODUCTION

In collaboration with Japanese researchers, we have been investigating the feasibility of high-temperature Rankine cycles using MHD (magnetohydrodynamic) generators for power conversion in future fusion power plants [1-7]. A cross-section view of a D-T tokamak reactor and the MHD power conversion system is shown in Figure 1, and details of a typical

outboard module in the equatorial plane of the reactor are shown in Figure 2. The blanket consists of about 480 such modules divided into twelve sectors with 40 modules in each sector. Vanadium alloy, which is compatible with the mercury working fluid, is used as the structural material. The mercury goes through the special Rankine cycle shown in Figure 3a [4,5].

The entire module shell is almost isothermal at a temperature somewhat above the selected evaporation temperature of the mercury coolant of 703 K. Mercury was chosen over an alkali metal because, at this temperature, T_s , the saturation pressure is only about 3.3 bar, which is much higher than the vapor pressures of the alkali metals. However, it is still low enough so that the module shell is subjected to relatively low pressure stresses. In addition, the line radiation loss from the mercury vapor is significantly less than for pure alkali metal vapors at our operating temperatures.

The blanket consists of a first wall zone, a low-temperature breeder zone, a high-temperature pebble-bed breeder zone and a high-temperature moderator/reflector pebble-bed zone. The mercury evaporates in the first two zones, and then is superheated in the two pebble-bed zones; additional superheat can be added if a microwave superheater (MWSH) is employed. The tokamak plasma parameters given in Table 1 have been optimized to maximize the production of synchrotron radiation for use primarily for current drive [8] and also for microwave superheating of the mercury vapor [3,6], which is seeded with a small amount of cesium to provide the ionization necessary for absorption of the microwaves. The resulting reactor has a reasonable neutron wall loading of 4.6 MW/sq.m.

The highly superheated vapor then flows into twelve MHD disk generators (one for each of the twelve sectors of the reactor) where the electric power is generated by direct conversion of the gas dynamic power into d.c. electric

power; the d.c. power is then inverted and sent to the grid as a.c. power. MHD generators have been chosen over conventional rotating turbine-generators for this advanced concept because of their compactness and high power density, which allows them to be placed inside the reactor vault. This can greatly reduce piping runs and capital costs. In addition, the MHD generators directly convert gas dynamic flow power into electric power in a volumetric process and thus require no moving parts to push on the hot gases; this allows them to work with higher temperature gases than conventional rotating turbines (for more information see e.g. Ref. 9).

Nonequilibrium MHD generators generally require inlet stagnation temperatures of 2000 K to 3000 K to give good performance, and hence are not suitable for the usual blankets proposed for fusion reactors in most of the systems studies to date. Our proposed blanket using vanadium alloy structure with ceramic pebble beds has the potential of producing these high stagnation temperatures at the outlet. MHD generators with the linear geometry have been proposed for use with fusion reactors on a few occasions in the past. For example, Rosa proposed using a linear cesium-seeded helium MHD generator with two high temperature blanket designs of Brookhaven in a Brayton cycle power plant [10]. A liquid metal MHD generator has also been proposed for use with gas cooled fusion reactors in Brayton power cycles [11]. Our proposed Rankine cycle concept has several potential advantages over a Brayton cycle power plant, including compact components with lower operating pressures and stresses, lower mass flow rates by using the enthalpy of evaporation, lower pumping power, very much reduced sensitivity to pressure drops in our CFAR type cycles, and good thermal efficiencies using superheating and internal regeneration [4]. These potential advantages ultimately should translate into lower capital costs and lower cost of

electricity.

There is no bottoming plant in this concept, and the entire power conversion system with the exception of the natural-draft dry cooling tower fits inside the reactor vault, as shown in Figure 1. The mercury condenser shown in Figure 1 is cooled by a suitable fluid such as the molten salt, HTS [12,13] which then flows outside the reactor vault to the cooling tower. The cooling tower will be very compact, since the mercury condensation temperatures (typically 480 to 500 K) and hence the heat rejection temperatures are high relative to ambient air temperatures.

In the following sections, the cycle will be described first and then the results of the thermal-hydraulic analysis of this first blanket concept will be summarized. Then the performance of an MHD disk generator suitable for this CFAR cycle will be discussed.

DESCRIPTION OF THE RANKINE CYCLE

The CFAR cycle shown in Figures 3a and 3b has several unconventional aspects designed either to minimize MHD pressure drops of the mercury blanket coolant or to increase the thermal efficiency. First of all the liquid mercury is pumped to a higher pressure, p_2 , than the desired saturation pressure, p_s . This is done so that after heating to saturation state 4, the liquid can be flashed into a mist flow at state 5 through specially designed orifices at the entrance to each module. This allows the mercury to be transported across the strong magnetic field with no magnetic pressure drop (as proposed in somewhat different form in Ref. [14]), and allows the entire module shell to operate at the low pressure, p_s , of about 3.3 bar. By overheating the vapor about 50 K above T_s at state 4, the flashing takes the mercury to a void fraction of over 90%, even though the flow quality is only

Figure 1a

Figure 16