CONTINUOUS TOKAMAK OPERATION WITH AN INTERNAL TRANSFORMER

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ABSTRACT

A large improvement in efficiency of current drive in a tokamak can be obtained using neutral beam injection to drive the current in a plasma which has low density and high resistivity. The current established under such conditions acts as the primary of a transformer to drive current in an ignited high-density plasma. In the context of a model of plasma confinement and fusion reactor costs, it is shown that such transformer action has substantial advantages over strict steady-state current drive. It is also shown that cycling plasma density and fusion power is essential for effective operation of an internal transformer cycle. Fusion power loading must be periodically reduced for intervals whose duration is comparable to the maximum of the particle confinement and thermal inertia timescales for plasma fueling and heating. The design of neutron absorption blankets which can tolerate reduced power loading for such short intervals is identified as a critical problem in the design of fusion power reactors.



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## 1. INTRODUCTION

The most efficient way to drive toroidal current in a tokamak is by electromagnetic induction using a set of external transformer coils. Because the current in the external coils cannot be ramped indefinitely, this method of current drive necessitates pulsed operation. The disadvantages of pulsed operation are discussed in recent reactor studies [1,2,3,4,5]. Severe problems are caused by interaction of the poloidal field (PF) coils which control current drive and plasma equilibrium and the toroidal field (TF) coils required for plasma stability. Pulsing the PF coil currents produces heating by eddy currents and also cyclic mechanical stresses due to interaction with TF coil currents. Cyclic mechanical stress can severely limit the number of pulses which can be applied to supporting structure, as illustrated schematically in Fig. 1. It is highly desirable to limit the cyclic portion of the stress to no more than about 10% of the total stress in a device which is to be pulsed up to one-hundred thousand times or more. In previous reactor designs with an external transformer, loss of plasma current and recharging the transformer coils leads to large cyclic stresses. As illustrated in Fig. 1, this might drastically limit the total allowable mechanical stress, and hence the total magnetic and plasma pressures, when compared to what might be obtained in a plasma with a strictly steady-state current driver. The purpose of this paper is to briefly review the problems with steady-state current drivers and then to show that a new method of current drive called the internal transformer cycle may avoid large cyclic stresses while driving a nearly constant plasma current at high efficiency.

The most straightforward way to overcome problems of pulsed operation is to drive the toroidal plasma current in a strictly steady-state device with externally supplied particles or waves. Previous studies have shown that such methods are feasible if an efficient external source can be found which drives at least 0.1 ampere of plasma current for every watt of power absorbed in a high density plasma [6]. Unfortunately, there are still questions whether such efficient current drive can be achieved. Radio frequency (RF) waves and relativistic electron beams (REB) can be generated efficiently but may not couple well to current drive in the center of the plasma. Prediction of current drive with these systems also requires nonlinear theories which may presently be too naive to apply to a turbulent plasma in a toroidal geometry. For neutral beams with energies up to 1 MeV/amu, the situation is reversed. Coupling current drive in the center of the plasma is simple and the theory of current drive is relatively straightforward. However, neutral beam power is expensive, particularly at higher energies where current drive is most efficient.

The difficulty of driving strictly steady-state current with neutral beams of more modest energy led one of the authors (CES) to propose a new method for current drive [4,5]. The aim is to avoid the major problems of pulsed operation while retaining much of the high efficiency of a transformer. Instead of reducing the parallel current in external coils, the electric field used to drive the plasma current inductively is provided by reducing a pre-established beam-driven current. Thus, the fast ions which carry the beam-driven current act as the primary of a transformer, just as if the plasma were threaded by a set of copper windings. Since the primary current of the transformer in this scheme lies in the plasma, we shall refer to this mechanism as "current drive with an internal transformer."

As with an external transformer, the primary current in an internal transformer cannot be steadily reduced forever. With an external transformer, restoring the primary current produces a negative loop voltage at the edge of the plasma; this induces an undesirable current that tends to disrupt the plasma (c.f., however, [7]). With an internal transformer, the effects of reestablishing the primary current are not so severe. This is because the primary current is in to the plasma and adds to the toroidal current, thereby maintaining stability of the current column.

It is important to control the plasma resistivity properly while driving the primary current of the internal transformer [4,5,8,9]. The reason this is true is as follows. A negative toroidal electric field is induced throughout the plasma when the primary internal transformer current is being reestablished. This negative electric field drives an undesirable electron current which cancels some of the primary transformer current. However, this electric field and the undesirable electron current decay at a rate

$$1/\tau_{skin} = \eta c^2/(4\pi L^2) ,$$

proportional to the toroidal resistivity $\eta$. (Here L is the scale height of variation across the plasma and c is the speed of light.) If the resistivity $\eta$ is kept high during the beam-driven phase of the transformer cycle, dissipation of the undesirable induced electron current can be very rapid. The most straightforward method of increasing the resistivity during the beam-driven phase of internal transformer operation is to add xenon or other high-Z material. Raising the resistivity during the current drive phase of internal transformer operation can be shown to increase the average current drive efficiency, albeit at the expense of briefly contaminating the plasma.

Another important advantage of the internal transformer over the external transformer is that the internal transformer method can require only small swings in the total toroidal current within the plasma. For example, one might allow the total current to decay for only 10% of the skin time $\tau^{c}_{skin}$ during the coast phase after shutting off the beam-driven current. Then enough beam-driven current would be added for a short fraction of the skin time $\tau^{d}_{skin}$ during the drive phase to re-establish the desired value of total plasma current. Thus, continuous operation of a tokamak plasma in this manner is possible with small variation in total plasma current. All that is required is that the coast phase be long compared with the particle confinement and thermal inertial times (e.g., so that impurities added during the drive phase may be flushed out) and short compared to the coast phase skin time. This requirement can be satisfied in a tokamak reactor where the heating and confinement times are expected to be of order of seconds and the skin time of order $10^3$ seconds.

Additional gains in current-drive efficiency may be realized by decreasing the plasma density during the beam-driven phase of the internal transformer operation [4,5]. This is particularly useful for beams of relatively low energy (120-200 keV) which have poor current-drive efficiency. But lowering the plasma density during current drive usually reduces the fusion wall loading and requires a somewhat lower vertical field. The tradeoff between these potential disadvantages and the increased efficiency of current drive is discussed below.

These methods of internal transformer operation originally suggested for use with neutral beam injection [4,5] can also be applied to other methods of current drive. For example, cycling plasma density was later discovered independently by Fisch, who outlined the scaling of current-drive efficiency for RF-driven reactors with density cycling in connection with studies of current drive by radio-frequency waves [10]. Cycled plasma resistivity arises naturally in the theory of current drive by relativistic electron beams. The first explicit suggestions that a large improvement in current driver performance may result from cycling plasma resistivity were made by Fisch [9], and Ehst et al. [8]. For detailed studies of both density and resistivity cycling, we will restrict ourselves to current driven by injection of neutral beams, since we have more confidence in the quantitative accuracy of the relevant theory [7,11,12] and its experimental confirmation [13,14]. If experiment shows agreement with theory for other methods of current drive, then the procedure presented here could readily be extended to these methods.

The problem we address in this paper is thus the optimization of plasma conditions for operation of an internal transformer driven by injection of neutral beams. We use a self-consistent description of the ion and electron temperatures in the coast and drive phases of transformer operation. We also calculate fusion power output, current-drive efficiency, resistive decay of induced currents, and the ratio of time spent in the coast and drive phases. We determine the efficiency of current drive and the plasma power multiplication factor Q averaged over a cycle of transformer operation. But since a previous study showed that neither of these two quantities is an adequate measure of reactor performance [6], we will concentrate on optimizing the relative cost of electrical generating capacity. We will compare the relative $/watt computed for various methods of current drive as a function of maximum allowed plasma pressure and cycling of heat loads. We can then give a brief outline of our estimate of where the optimal reactor design may actually lie. We believe it will be evident that significant reductions in the overall problems of current drive should be achievable using an internal transformer, and that this option should be seriously considered in future reactor designs.

## 2. COMPUTATIONAL MODEL

We now describe the equations we solve to determine self-consistent plasma parameters, current-drive efficiency, and relative cost of electrical generating capacity. We then state how constraints on cycling of fusion power are applied, and which parameters are varied during a single optimization of reactor cost. The plasma parameters include ion and electron temperatures determined by power balances which include various possible anomalous electron and ion losses as well as ion energy losses due to toroidal field ripple, as described in Section 2.1. Current-drive efficiency is determined by generalizing our previous analytic approximation from the case of strict steady state [6] to the present case of an internal transformer cycle, as described in Section 2.2. Our model for computing the relative cost of electrical generating capacity includes fixed costs proportional to the engineered reactor volume cost of installed neutral beam power, and cost of converting thermal to electrical power, as in our previous work [6]. The simple generalization of our previous model required to treat a transformer cycle is described in Section 2.3. A complete and precise description of the computational model is available elsewhere [15].

### 2.1. Power Balances

We solve steady-state, zero-dimensional (0-d) electron and ion power balance equations using recently derived scalings for transport due to toroidal field ripple [16] and anomalous transport at low [17] and high [18] plasma pressure. Steady state is adequate because the coast and drive phases are assumed long compared to an energy confinement time. 0-d is required by the complexity of our multidimensional parameter search, but the fusion power production and current-drive efficiency are normalized to profiles from the

1-d BALDUR transport code and the ion orbit code IO, as in our previous paper [6].

The energy balance equations are

$$P_{\alpha e} + P_{be} = E_e/\tau_{Ee} - Q_\Delta + P_{vol,e} ,$$

$$P_{\alpha i} + P_{bi} = E_i/\tau_{Ei} + Q_\Delta .$$

For each species j, $P_{\alpha j}$ and $P_{bj}$ are heating by fast $\alpha$ and beam particles, and $E_j = 1.5 n_j T_j$ and $\tau_{Ej} = L^2/\chi_j$ are the energy content and confinement time. The volume losses $P_{vol,e}$ are due to coronal equilibrium radiation. $Q_\Delta$ is the heating of ions by collisional energy interchange. The formulas for $P_{vol,e}$ are taken from Post et al. [19] and the remaining terms (except for $\tau_{Ej}$ described below) are defined in Mikkelsen and Singer [6]. Unless otherwise noted, all formulas in this paper are in Gaussian units.

To estimate confinement losses, we set the plasma radial scale length $L = (ab/4)^{1/2}$ where a is the plasma half-width and $b \equiv \kappa a$ is the half-height. When applying formulas for the thermal diffisivities $\chi_j$, we treat the plasma as an equivalent circle with minor radius $a_{eq} = \kappa^{1/2} a$. The ion thermal diffusivity $\chi_i$ is an approximation [15] to the formulas of Shiang and Callen [10] and includes their recently derived scaling for the transition from collisionless through collision-dominated transport. To account for convective transport of the ion energy, we add $5 \times 10^{16}/n_e$ to $\chi_i$, since this value is roughly consistent with the results of the detailed transport simulations discussed in Ref. 6. Negligible losses due to toroidally symmetric neoclassical processes are ignored. The electron diffusivity $\chi_e = 5 \times 10^{17}/n_e + \chi_\beta$ includes INTOR scaling and an additional contribution $\chi_\beta$ which dominates at high plasma pressure. Two different forms are used for $\chi_\beta$. A standard form

$$\chi_\beta = \begin{cases} (5 \times 10^{17}/n_e)\{\exp[f_\beta(\beta-\beta_{crit})] - 1\}, & \beta > \beta_{crit} \\ 0, & \beta < \beta_{crit} \end{cases} ,$$

is motivated by ideal ballooning mode theory which predicts onset of instability when $\beta = (2/3)(E_i + E_e)/[B^2_{toroidal}/(8\pi)]$ exceeds some critical value $\beta_{max}$ (or, equivalently for our purposes, when $\beta_p = (2/3)(E_i + E_e)/[B^2_{poloidal}/(8\pi)]$ exceeds some critical value). In this form of $\chi_\beta$, we generally use $f_\beta = 10^3$ to give a "hard" $\beta$ limit. An alternate "soft" $\beta$ limit form is one of a series of possible scalings derived by Sigmar and Houlberg to fit data from ISX-B following concepts from resistive MHD stability theory [17],

$$\chi_{\beta p} = \chi_*(\varepsilon\beta_p/\varepsilon_*)^{n_{\beta p}} ,$$

where $\varepsilon = a_{eq}/R$ is the inverse aspect ratio. We shall choose reference values $\chi^*, \varepsilon^*$ and $n_{\beta p}$ found by Sigmar and Houlberg to be roughly consistent with results from the ISX-B experiment.

## 2.2. Current Drive

We need to know the proportion of the total transformer cycle of length $\tau$ which is spent in a coast phase of length $\tau_c$ and a drive phase of length $\tau_d$. To do this we approximate the rate of change of current in the coast phase by

$$\frac{|\Delta I|}{\tau_c} \approx \frac{I_{av}}{\tau^c_{skin}} ,$$

and in the drive phase by

$$\frac{\Delta I}{\tau_d} \approx \frac{I_{bd} - I_{av}}{\tau^d_{skin}} .$$

Here $\Delta I$ is the total current swing, $I_{av}$ is the average toroidal current, and $I_{bd}$ is the beam-driven current (the current which would eventually be obtained if neutral injection could be continued for many skin times). Dividing the above equations shows that the coast/drive ratio

$$\frac{\tau_c}{\tau_d} = \frac{\tau^c_{skin}}{\tau^d_{skin}} \left(\frac{I_{bd}}{I_{av}} - 1\right) ,$$

is equal to the ratio of the skin times multiplied by the "overdrive"

$$O_{drive} = \frac{I_{bd}}{I_{av}} - 1 .$$

This result was first derived in a convenient form by Fisch [10].

The beam-driven current $I_{bd}$ is determined by [6]

$$I_{bd} = A_{bd} P_{abs} \frac{(T_e/T'_e)\, J(x,y)/0.2}{(R/R')(n_e/n'_e)} \{Z_b^{-1} - Z_{eff}^{-1}[1-G\,(\epsilon,Z_{eff})]\}\cos\theta_{inj} ,$$

where $P_{abs}$ is the beam power absorbed in the plasma, $(R/R')$ is the plasma major radius divided by 500 cm, $(T_e/T'_e)$ is the electron temperature divided by $10^4$ eV, and $(n_e/n'_e)$ is the density divided by $10^{14}\,cm^{-3}$. $J(x,y)$ is a dimensionless function of $T_e$ and beam energy $E_b$ and charge $Z_b$ and of species masses and charges and concentrations with a value typically of $J(x,y) \lesssim 0.2$ [6]. G is the banana-regime limit of the trapped electron correction to the collision electron return current and is a function of inverse aspect ratio and effective charge $Z_{eff}$ [12,6]. $\theta_{inj}$ is the angle between the beam centerline and the magnetic axis of the plasma. $A_{bd}$ is a normalization constant which we determine using the ion orbit code, IO, run with plasma profiles generated with the BALDUR 1-d transport code in the manner described in a previous work [6]. The standard value of $A_{bd}$ used in this paper is $A_{bd}$ = 37.2 statamp/(erg/s) = 0.124 amp/watt. The sensitivity of our results to this current-drive efficiency coefficient is discussed below.

For comparison with earlier studies, we define two parameters which have often been taken to be measures of performance of steady-state reactors. The average current-drive efficiency in amps/watt is

$$\eta_{drive} = (10^6/c)\ I_{av}/P_{b,av} .$$

Here $(10^6/c)$ is a constant to convert from Gaussian units to amps/watt, and

$$P_{b,av} = \frac{\tau_d}{\tau_d + \tau_c} P_{bd}$$

is the applied beam power averaged over the transformer cycle (assuming no beam power is applied during the coast phase). We also define the plasma power multiplication factor

$$Q_{av} = 5\ \frac{\tau_c P_\alpha^c + \tau_d P_\alpha^d}{(\tau_c + \tau_d)\ P_{b,av}} ,$$

where $P_\alpha^c$ and $P_\alpha^d$ are the fusion alpha power produced in the coast and drive phases.

## 2.3. Costing

In our previous study [6], we found that $\eta_{drive}$ and $Q_{av}$ are not adequate measures of reactor performance, so we also modify the cost model in that study to include operation of the internal transformer. The relative capital cost becomes

$$C = C_V + C_b P_b + C_{th} P_{th,av} ,$$

where $C_V$ is the volume-related cost, $C_b P_b$ is the cost of installed beam power $P_b$,

$$P_{th,av} = \frac{5 f_{bl} P_\alpha^c \tau_c + (5 f_{bl} P_\alpha^d + P_b)\, \tau_d}{\tau_c + \tau_d} ,$$

where $C_{the} = C_{th} P_{th,av}$, $C_{th}$ is the cost per unit power of converting thermal to electrical power, and the average thermal power. Here the power multiplication factor $f_{bl}$ is related to the neutron multiplication $M_b$ [6] by the formula

$$f_{bl} = (4M_b + 1)/5 .$$

One unit of relative cost in our study corresponds to the total direct cost (in 1981 dollars) of equipment and buildings specific to the tokamak, power generation, or neutral beam systems, including allowances for design, contingencies, and spare parts [6]. Indirect costs and direct costs which scale with the cost of the whole system are not included. The average electrical power output is

$$P_e = \eta_e P_{th,av} - P_V - P_{b,av}/\eta_b ,$$

where $\eta_e$ is the net efficiency of the generating system and $P_V = P_E V_E$ is a power drain proportional to the engineered volume $V_E$. $\eta_b$ is the net efficiency of producing neutral beam power from electricity (after accounting for any recovery of power from the beamlines). Parameters of the costing model for the tokamak and power generation are averages of those from the NUWMAK and STARFIRE studies [1,2]. Neutral beam parameters from printed [20,21] and private communications [22] are estimated using the same units of relative cost. We have implicitly assumed that the system has enough thermal or mechanical inertia to produce a constant power output from the turbines and a constant power drain to run the beamlines. For a drive time of a few seconds, this would appear not to be an unduly restrictive assumption [1].

### 2.4. Constraints and Optimization

As mentioned above, we sometimes constrain the cyclic change in heat load when optimizing performance of given reactor. We set

$$P_\alpha^d = f_{swing} P_\alpha^c ,$$

where $f_{swing}$ relates the fusion power in the coast and drive phases.

To optimize plasma parameters for a given set of machine parameters and constraints, we do a five-dimensional parameter search. The two coast phase parameters varied are electron density $n_e^c$ and toroidal beta $\beta_c$. Given $n_e^c$ and $\beta_c$, the electron power balance in the coast phase determines the coast electron and ion temperatures. The ion energy balance is then inverted to find the corresponding toroidal field ripple amplitude $\delta$. Three parameters are varied in the drive phase. These are the toroidal beta $\beta_d$, the effective charge $Z_d$, and the overdrive $O_d$ defined above in Section 2.3. The electron and ion power and balances in the drive phase are used to find the temperatures $T_e^d$ and $T_i^d$. The only solutions considered when searching for optimum plasma parameters are those which satisfy the ion and electron energy balances in both phases of the transformer cycle and all of the other constraints mentioned above.

## 3. RESULTS

First, we give results for a reference case. Then we vary $\beta_{max}$, the constraints on cycling fusion power, the beamline parameters, the machine size, and the transport assumptions.

### 3.1. Reference Case

As a point of reference, we consider a machine in the size range of NUWMAK [1], INTOR [2], and a recent DEMO design [23]. The parameters for this case are listed in Table 1. Results of minimizing the cost of generating capacity are listed in regular type. The other parameters listed in italics are constant throughout this paper except where variations are noted.

For the reference case, the required beam power is kept relatively small by taking advantage of increased current-drive efficiency in a low density drive phase. Increasing the resistivity in the drive phase reduces the average power needed to drive the current. The relative importance of these two mechanisms and their utility for reducing the cost of an internal transformer system depends on the achievable plasma pressure, the allowed cycling of the fusion power, the neutral beam energy and efficiency, and the size of the tokamak.

3.2. $\beta_{crit}$

In the reference case, electron energy confinement degrades rapidly for $\beta > \beta_{crit} = 0.067$. Results for other values of $\beta_{crit}$ are shown in Fig. 2; the cross locates the reference case.

The benefits of the internal transformer can be seen by comparison to results for strict steady state (dashed line in Fig. 2). Use of the internal transformer is especially important at lower $\beta$'s, where most of the fusion power output would be used to drive the current in a reactor operating with time-invariant plasma parameters.

More insight into the savings effected by using an internal transformer can be obtained by comparing the results for an idealized pulsed reactor with "free" current drive. The lower dot-dashed line in Fig. 2 shows the results for pulsed ignition reactor driven by an idealized external transformer. To obtain these results we simply maximized the fusion power in the reference model without requiring current drive. Evidently the internal transformer is much closer to this idealized case than to strict steady state.

A more realistic comparison with the traditional external transformer is given by the upper dot-dashed line in Fig. 2. For this case, we reduced the toroidal field by 20% as an example of what might be necessary to compensate for mechanical fatigue due to pulsing an external transformer. We also impose a small charge for an engineered OH coil region of area $7m^2$ and half-height 5.62m. An increment of 10% is added to the cost to account for the typical duty factor in this size reactor [1]. We believe these assumptions are an optimistic model for a tokamak reactor with a pulsed external transformer. Thus, the internal transformer is the preferred mode of current drive if the parameters listed in Table 1 are achievable.

### 3.3. Fusion Power Loading

The constraints on the internal transformer cycling used above may be too lenient. In particular, designing a structure to take short periods of reduced power from the plasma might significantly increase the cost of a reactor system. If so, less flexibility is allowed in optimizing the internal transformer. Results for various drive/coast power load ratios are shown in Fig. 3. If the power load must be approximately constant over the transformer cycle ($f_{swing} \sim 1$), the performance of the internal transformer is significantly degraded (c.f. discussion in Section 4).

The desirability of maintaining a constant power load has recently been pointed out by Fisch in connection with a discussion of oscillating resistivity and current drive by radio-frequency (RF) waves and beams [9]. We chose our reference case ($E_b$ = 400 keV, $\eta_b$ = 60%, $C_b$ = \$1/watt with \$ defined as in Section 2.4) to give a performance similar to what could be hoped for from the better methods of RF current drive, and we were unable to gain significant benefit from oscillating resistivity while keeping constant fusion power loading. This is because of the electric power and capital cost required to provide the significant current overdrive which is needed when cycling plasma resistivity while maintaining constant fusion power. We believe that RF current drive is unlikely to give performance sufficiently better than our reference case or that resistivity cycling alone will prove to have significant advantages over strictly steady-state current drive.

### 3.4. Beam Parameters

A major uncertainty for current drive in tokamaks is the performance of various drivers. Results from varying the beam cost and efficiency for an internal transformer driven by beams of various energies are shown in Fig. 4. Nominal efficiencies and energies of three different beam systems are noted in Fig. 4. The reference case described in Table 1 corresponds to the cross in Fig. 4a. This probably represents the lower limit of capital cost of a neutral deuterium beam and is also representative of what might be obtained with good performance of some methods of radio-frequency current drive. The case labelled EA corresponds to neutralization of electrostatically accelerated $D^-$ ions; by comparison, this case gave a reactor cost of 6.3 $w_e^{-1}$ in our previous study of steady-state reactors [6]. The case labelled RFQ corresponds to neutralization of $D^-$ ions accelerated to 400 keV in a radio-frequency quadrupole accelerator, the feasibility of which has already been demonstrated [22]. Higher beam energy [6] would be highly desirable for this case, but is of modest help for the EA case. A beam energy of 400 keV is evidently sufficient to drive a fusion power reactor of modest size with a cost comparable to the 2.5 $w_e^{-1}$ computed for "free" current drive in the context of our model.

Figure 4b shows that using a more accessible beam energy of 200 keV is adequate for a "demonstration reactor" where the cost of power produced is not the primary consideration. However, beams of this energy do not drive current efficiently and force a higher operating density due to excessive beam fueling, so they are not attractive for power production reactors. A demonstration reactor could be driven by even more inefficient and expensive beams from $D^+$ sources, but such a reactor would give much better performance if higher energy beam derived from $D^-$ sources were developed.

Current drive with the full energy fraction from beams with the "state-of-the-art" energy of 120 keV is too inefficient to allow sufficient margin for use in a demonstration reactor, as illustrated by the large costs plotted in Fig. 4c. However, such beams would be adequate for a device designed as a continuously operating source of 14 MeV neutrons. Thus, the initial stage of operation of a Fusion Engineering Device, or of the INTOR reactor, could reduce problems associated with current cycling by incorporating internal transformer operation driven by neutral beams of modest energy. Later stages of operation of these devices could improve performance if more efficient current drive became available.

### 3.5. Machine Size

The size of our reference reactor was chosen in a range where most recent design studies have provided a conceptual frame of reference. To assess the ultimate impact of the internal transformer concept on power production, we have also computed results for a machine the size of the STARFIRE power reactor [3]. The machine parameters used were R = 7 m, a = 1.94m, and elongation $\kappa = 1.6$. The engineered volume was $V_E = 5150m^3$. The confinement time of the injected deuterons was scaled in proportion to $a^2$ from the reference case value of $\tau_p = 1.0s$ to give $\tau_p = 2.6s$. The plasma current was increased to 12 MA to maintain the same safety factor q as in our reference case. All other input parameters were those in Table 1. For the reference beams the relative costs of generating capacity were 2.3:1.4:1.3 when comparing strict steady state: internal transformer: idealized external transformer. Thus, in the context of our model, current drive with an internal transformer is "nearly free" in this 2 $GW_e$ reactor, and the cost of electricity is smaller by a factor of two than in the "demonstration reactor" denoted by the cross in Fig. 2.

3.6. Transport Assumptions

The results quoted above are relatively insensitive to the assumptions we made about energy confinement. This is true because there is generally excess fusion power available in the coast phase of the internal transformer cycle and excess fusion or beam heating power available in the drive phase. In the coast phase, operation at low density insures high electron temperature even with relatively poor confinement. The drive phase requires only sufficient electron energy confinement to avoid excessive electron drag on the circulating fast ions.

As an example of an alternate confinement scaling which may have a sounder basis in theory and experiment than our simple reference model, we add to the electron thermal diffusivity a contribution of the type described above in Section 2.2

$$\chi_{\beta p} = 2000 \, \left[(a_{eq} R)\beta_p/\varepsilon_*\right]^{n_{\beta p}} ,$$

where $a_{eq} = \kappa^{1/2} a$ is an equivalent circle radius and R is the major radius of the magnetic axis. With a scaling exponent $n_{\beta p} = 2$, this addition makes our energy confinement consistent with results at high values of $\beta_p$ in ISX-B experiments with an inverse aspect ratio of $\varepsilon_* = 0.29$. This modification of $\chi_e$ precludes ignition in our reference INTOR-sized plasma; but it allows ignition in the STARFIRE power reactor described in Section 3.5 and increases the cost of electricity in our model by only one percent for this machine. In keeping with the spirit of the $\chi_{\beta p}$ scaling, however, we should also remove the rapid decrease in energy confinement above $\beta_{crit}$ when adding the $\chi_{\beta p}$ scaling. Doing so still leaves ignition precluded in our INTOR-sized plasma but gives improved performance in our STARFIRE power reactor model for a $\beta_p$

scaling exponent of $n_{\beta p} \lesssim 2.5$. It should also be noted that thermal stability would be a problem in this model for $n_{\beta p} < 2$. The implication of these results is simple. The constraints on electron energy confinement required for effective operation with an internal transformer in the coast phase are primarily that it be sufficient to approach ignition in a thermally stable manner. These constraints are not qualitatively different from those that apply to any pure fusion power reactor.

The rather extreme ion/electron temperature ratio listed for the drive phase in Table 1 is not essential for successful operation of an internal transformer. This is evidenced by the fact that doubling the anomalous ion thermal diffusivity or increasing the ripple-trapping and/or banana-drift ion thermal diffusivities by an order of magnitude gives less than 1% increase in the minimized cost of electricity. Even the rather extreme step of interchanging the anomalous ion and electron thermal diffusivities in the reference case described above only increases the minimized cost of electricity by 20%. Half of this increase is because the lower ion temperature of $T_i^c = 7.3$ keV in the optimized coast phase gives lower fusion power output, and the other half of the increase results from the larger skin time that accompanies the increased electron temperature of $T_e^d = 19$ keV in the drive phase.

The particle confinement time may be an important consideration for the drive phase of an internal transformer. This is the case because fueling of the plasma by neutral injection may unduly increase the plasma density and increase the drag on circulating fast ions. We therefore tried doubling the global particle confinement time to $\tau_p = 2$ sec in our reference case. This raised the density in the drive phase to $n_d = 0.8 \times 10^{13} cm^{-3}$ and increased the cost of electricity by 3%. Although a central particle confinement time of $\tau_p < 2$ sec seems reasonable for a density of $n_d \sim 10^{13} cm^{-3}$, it should be kept in mind that some control over particle recycling may also be necessary to achieve this low density. A crude model of particle recycling [24] suggests that on the order of one out of every one-hundred recycling deuterons must be removed during the drive phase to avoid having recycling particles dominate the beam fueling.

We have also varied the current-drive efficiency in our model in order to test the sensitivity of the results to this parameter. We did this primarily to gain insight into what would happen if the plasma parameters in the drive phase allowed lower current-drive efficiency than in our model. For example, an attempt to minimize thermal cycling of the blanket might require a drive phase which is not long compared to the plasma confinement and thermal inertial timescales. In this case, the plasma density approaches its minimum value for only a fraction of the drive phase. A rough idea of what this implies for internal transformer action is given by the result that halving the current-drive efficiency in our reference case increased the minimized cost of electricity by 5%. A more careful assessment of this problem would require time-dependent transport code simulations based on sound extrapolations of confinement data, a project beyond the scope of this paper.

To summarize the internal transformer method of current drive is insensitive to all reasonable variations in confinement scaling, provided an acceptable thermal equilibrium exists in the coast phase. For a relatively efficient driver, such as 400 keV neutral beams, there is a substantial margin of safety to increases in particle confinement and decreases in current drive efficiency compared to the assumptions in our reference case.

4. DISCUSSIONS AND CONCLUSIONS

The internal transformer concept combines some of the major advantages of steady-state operation with the efficiency of inductive current drive. These advantages include elimination of large external transformer coils from the valuable space near the center of the machine and may allow elimination of large cyclic stresses which would otherwise result from cycling external transformer coils. (Reduction of cyclic stresses is particularly important as there is at present no careful treatment of the related mechanical fatigue problem which shows that external transformer cycling is compatible with economical operation of a pure fusion power reactor.) The current-drive efficiency of an internal transformer is sufficiently good that the cost of a reactor with an internal transformer generally comes closer to "free" current drive than for one with strictly steady-state current drive in the context of our computational model. In the likely event that power from a strictly steady-state reactor is in turn cheaper than a realistic pulsed ignition device, the internal transformer is clearly the preferred method of current drive.

However, there are two problems not addressed in our model which could conceivably compromise the choice of an internal transformer for current drive. The first problem is cycling of the fusion power loading, which is essential for optimal performance in our model. Although this causes problems particularly with design of the reactor blanket, there are two reasons why the internal transformer is likely to remain in the preferred design. One reason is that sufficient thermal inertia might be incorporated in the blanket to mitigate the thermal cycling problem. An extreme example of this is given by the NUWMAK design [1]. If the drive phase of each internal transformer cycle can be kept to ten seconds or less, then cycling of fusion power may not be too serious a problem. Another consideration is that so-called strict steady-state reactors will in any case have to deal with some degree of cycling of the fusion power. This could result either from occasional ingestion of impurity flakes or dust from the limiter or wall, or it could result from periodic readjustment of the plasma current and temperature profiles (sometimes called "giant sawteeth"). In a large power reactor, giant sawteeth could cause 10-20% fluctuations of the fusion power with a repetition period of seconds. Thus variable fusion power loading may be an endemic problem of tokamak reactors, and its presence cannot be used to categorically exclude a certain operation mode without a careful analysis of the problems it creates in specific designs.

A second problem not addressed in our model is cycling of the vertical field. For example, using a simple equivalent circle model for the vertical field of $B_V = (I_{av}/R)[\ln(8R/A_{eq}) + \beta_p + (\ell_i - 3)/2]$ where $\ell_i \approx 0.5$ is the internal inductance, the vertical field varies between $B_V = 2.6T$ and $B_V = 5.9T$ in the optimized internal transformer cycle illustrated in Table 1. The concommitant change in poloidal field coil currents is small compared to that which result from cycling the current in external transformer coils, but the design consequences of cycling the vertical field could still be significant. However, the interaction of cyclic vertical fields with toroidal field coils could be minimized by placing small copper equilibrium field coils inside the bore of the toroidal field coils. These copper coils would only carry current during the relatively short drive phase of the transformer cycle. A steady-state current would be carried in external superconducting coils to produce sufficient vertical field for the high $\beta_p$ coast phase of the transformer cycle.

Choice of the appropriate current driver for an internal transformer should occupy a significant amount of the work on tokamak design in the next few years. First our analysis should be repeated using existing models of RF and possibly of REB drive. Improvements of RF and REB theory and of the basis for extrapolating neutral beam technology are also needed. Experimental tests of the internal transformer with various current drivers are highly desirable. Such studies should clarify whether other current drivers could give even better performance than neutral beams of moderately high energy.

The internal transformer also has profound implications for the design of the next round of tokamak experiments and engineering studies. The design goal motivating these investigations will be to show that physics and technology are available to proceed with construction of a demonstration reactor. If a demonstration reactor is to use an internal transformer, then advanced tokamaks should also incorporate this technology. But until now the technology for a current drive in a device with $R \lesssim 5m$ appeared not to be available in the necessary time frame. This lead to the paradoxical situation where the latest power and demonstration reactor designs assume steady state, while INTOR and other advanced tokamaks use a pulsed external transformer. The greatly increased efficiency of the internal transformer should allow INTOR and similar devices to achieve continuous operation with current drivers of modest efficiency. Coupled with adoption of the internal transformer in DEMO designs, this would resolve the present paradox.

The key question in designing a demonstration power reactor with an internal transformer is the impact of a variable fusion power output on the first wall and blanket design. There are two reasons why this question must be addressed. First, as we have illustrated in Fig. 3, cycling plasma resistivity without cycling fusion power output is unlikely to provide significant improvements over the relatively poor performance of strictly steady-state pure fusion tokamak reactors. Second, the minimum length of the drive phase of an internal transformer is likely to be several seconds, which may be comparable to the thermal inertia time constant of some blanket components in some types of blanket design. This is what makes the choice of a blanket design compatible with a realistic scenario for current drive into a key question for fusion reaction design.

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TABLE 1. Reference Case

| PARAMETER | COAST | DRIVE | AVERAGE |
|---|---|---|---|
| 1. Machine Dimensions | | | |
| $V_E$ = Engineered Volume ($m^3$) | - | - | 3600 |
| A = Wall Surface Area ($m^2$) | - | - | 356 |
| 2. Plasma and Neutral Beam Parameters | | | |
| R = Major Radius (m) | 5.3 | 5.3 | 5.3 |
| a = Half-width (m) | 1.2 | 1.2 | 1.2 |
| $\kappa$ = Ellipticity | 1.6 | 1.6 | 1.6 |
| $n_e$ = Electron Density ($10^{14} cm^{-3}$) | 3.1 | 0.048 | 3.0 |
| $\beta$ = Toroidal $\beta$ (%) | 6.9 | 0.22 | 6.7 |
| $\beta p$ = Poloidal $\beta$ | 2.9 | 0.094 | 2.8 |
| $B_V$ = Vertical Field (T) | 6.0 | 2.6 | 5.9 |
| I = Plasma Current (MA) | 5.8-7.0 | 5.8-7.0 | 6.4 |
| $T_i$ = Ion Temperature (keV) | 8.7 | 42.0 | 9.7 |
| $T_e$ = Electron Temperature (keV) | 8.4 | 6.0 | 8.3 |
| $Z_{eff}$ | 1.2 | 16. | 1.7 |
| Plasma Species $f_a = n_a/n_e$ | | | |
| $f_D$ | 0.45 | 0.62 | 0.46 |
| $f_T$ | 0.45 | 0.003 | 0.44 |
| $f_{He}$ | 0.050 | 0.050 | 0.050 |
| $f_{Xe}$ | 0.000035 | 0.0052 | 0.00019 |
| $\tau$ = Pulse Length (s) | 45. | 3.5 | - |
| $\tau_{skin}$ = Skin Time (s) | 220 | 17. | 210 |
| $\tau_p$ = $D^+$ Confinement Time (s) | - | 1.0 | - |
| $P_b$ = Beam Neutral Power (MW) | - | 60. | 1.8 |
| $E_b$ = Beam Energy (MeV) | - | 0.40 | - |
| $R_{tang}$ = Beam Tangency Radius (m) | - | 5.1 | - |
| $\eta_b$ = Electric → Beam Efficiency | - | 0.60 | - |
| Q = Plasma Energy Gain | $\infty$ | 0 | 880 |

TABLE 1 (continued)

| PARAMETER | COAST | DRIVE | AVERAGE |
|---|---|---|---|
| **3. Power Output** | | | |
| $5P_\alpha$ = Fusion Power (MW) | 1650 | 0.0 | 1600 |
| $M_b$ = Blanket Neutron Amplification | 1.2 | 1.2 | 1.2 |
| $P_e$ = Gross Electrical Output ($MW_e$) | - | - | 640 |
| $\eta_e$ = Thermal → Electric Efficiency | 0.35 | 0.35 | 0.35 |
| $P_V$ = Volume Dependent Losses ($MW_e$) | 72. | 72. | 72. |
| $P_{net}$ = Net Electrical Output ($MW_e$) | - | - | 560 |
| $I_W$ = Wall Life ($MW\text{-}yr/m^2$) | - | - | 10. |
| $t_{down}$ = Time to Repair Neutron Damage (yr) | - | - | 0.50 |
| $t_{up}$ = Time between Damage Repairs (yr) | - | - | 2.8 |
| $P_{net}^{av}$ = Average Electrical Output ($MW_e$) | - | - | 470 |
| **4. Reactor Components** | | | |
| $P_W$ = "First Wall" Loading ($MW/m^2$) | 0.93 | 0.17 | 0.90 |
| $P_n$ = Neutron Wall Loading ($MW/m^2$) | 3.7 | 0.0 | 3.6 |
| $B$ = Field on Axis (T) | 5.5 | 5.5 | 5.5 |
| $N_{coil}$ = Number of TF Magnets | 12 | 12 | 12 |
| $\delta$ = Volume Average Ripple (%) | 0.13 | 0.13 | 0.13 |
| **5. Cost Model** | | | |
| $C_E$ = Cost/Engineering Volume ($cm^{-3}$) | - | - | 0.28 |
| $C_{th}$ = Thermal → Electric Cost ($w^{-1}$) | - | - | 0.11 |
| $C_b$ = Cost of Installed Beam Power ($w^{-1}$) | - | - | 1.0 |
| $C_E V_E$ = Cost of Engineered Volume (M) | - | - | 1010 |
| $C_{the}$ = Cost of Thermal→Electric Equipment (M) | - | - | 200 |
| $C_b^{Pb}$ = Cost of Beams (M) | - | - | 60 |
| $C$ = Cost of Electricity Production ($w_e^{-1}$) | - | - | 2.7 |

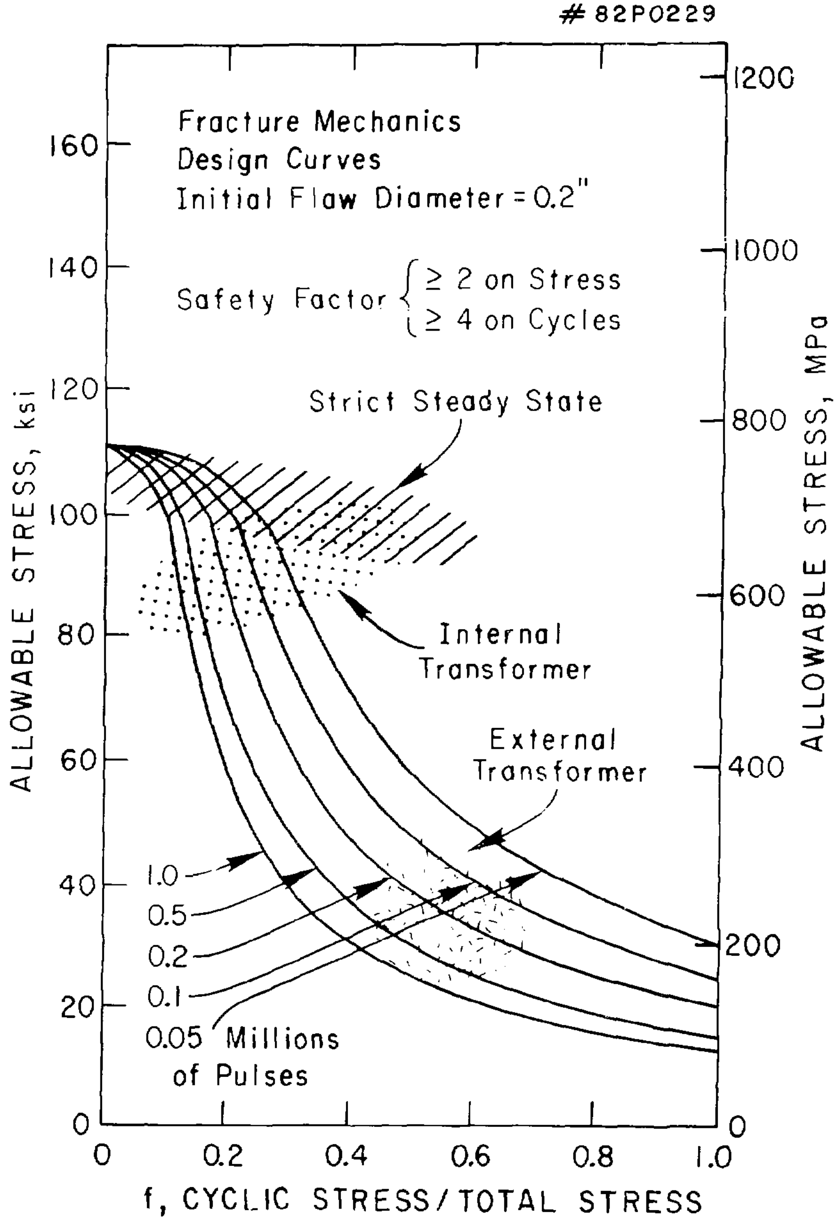
FIGURE CAPTIONS

Fig. 1. Schematic illustration of conceptual stress cycling regimes for various types of power reactor (adapted from Spampinato et al. [25]).

Fig. 2. Relative cost of electricity vs. critical $\beta$ for various current drive options. Cross (x) denotes reference case, Table 1.

Fig. 3. Relative cost of electricity for reference case tokamak transport model with various limits on fusion power cycling $f_{swing} = P_{\alpha}^{d}/P_{\alpha}^{c}$. Upper and lower arrows are costs at $\beta_{crit} = 0.067$ from Fig. 2.

Fig. 4. Contours of constant relative cost of electricity (in $w_e^{-1}$) vs. beam efficiency $\eta_b$ and relative capital cost $C_b$ of power in beam neutrals for beam energies characteristic of (a) a power production reactor, (b) a demonstration reactor, and (c) a near-term neutron source. Ovals represent possible ranges of beam parameters, and symbols therein denote examples of nominal beam systems mentioned in text.



# 82P0229
Fracture Mechanics
Design Curves
Initial Flaw Diameter = 0.2"
Safety Factor ≥ 2 on Stress
≥ 4 on Cycles
Strict Steady State
Internal Transformer
External Transformer
1.0
0.5
0.2
0.1
0.05 Millions of Pulses
ALLOWABLE STRESS, ksi
ALLOWABLE STRESS, MPa
f, CYCLIC STRESS/TOTAL STRESS
0
20
40
60
80
100
120
140
160
0.2
0.4
0.6
0.8
1.0
200
400
600
800
1000
1200


Fig. 1

#82P0202

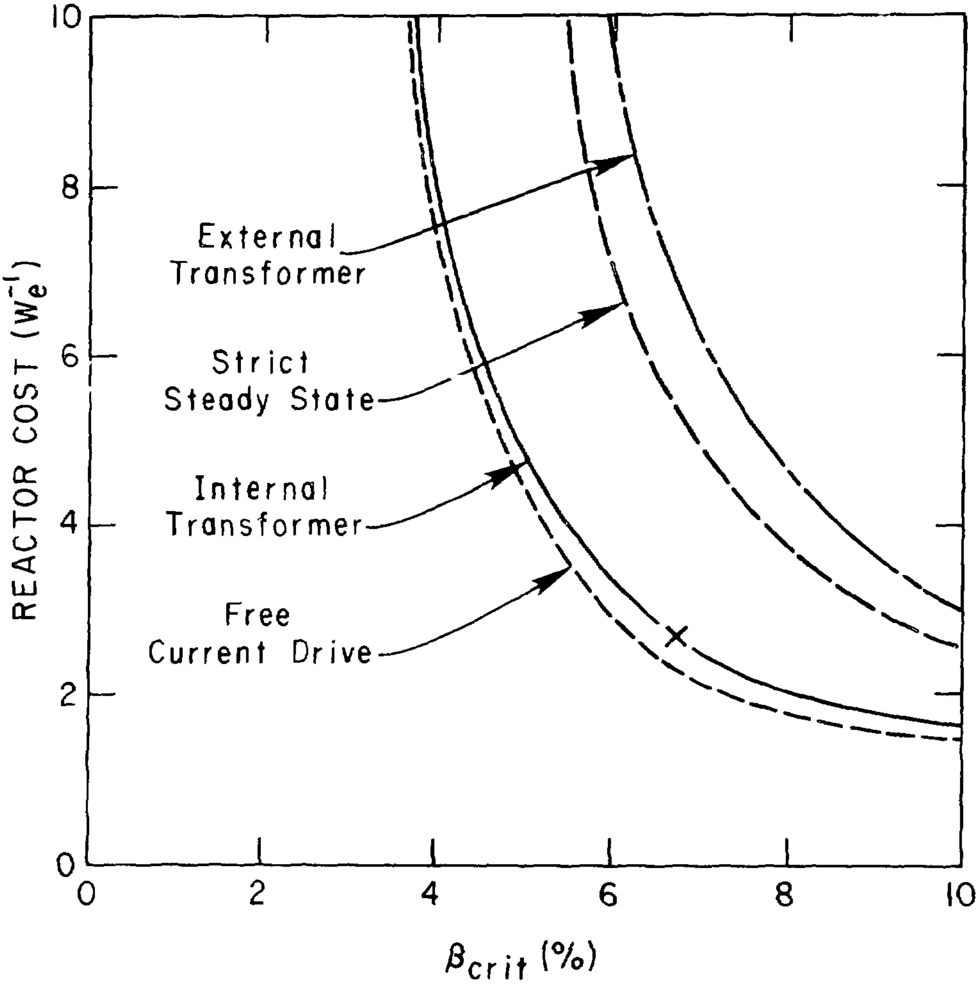
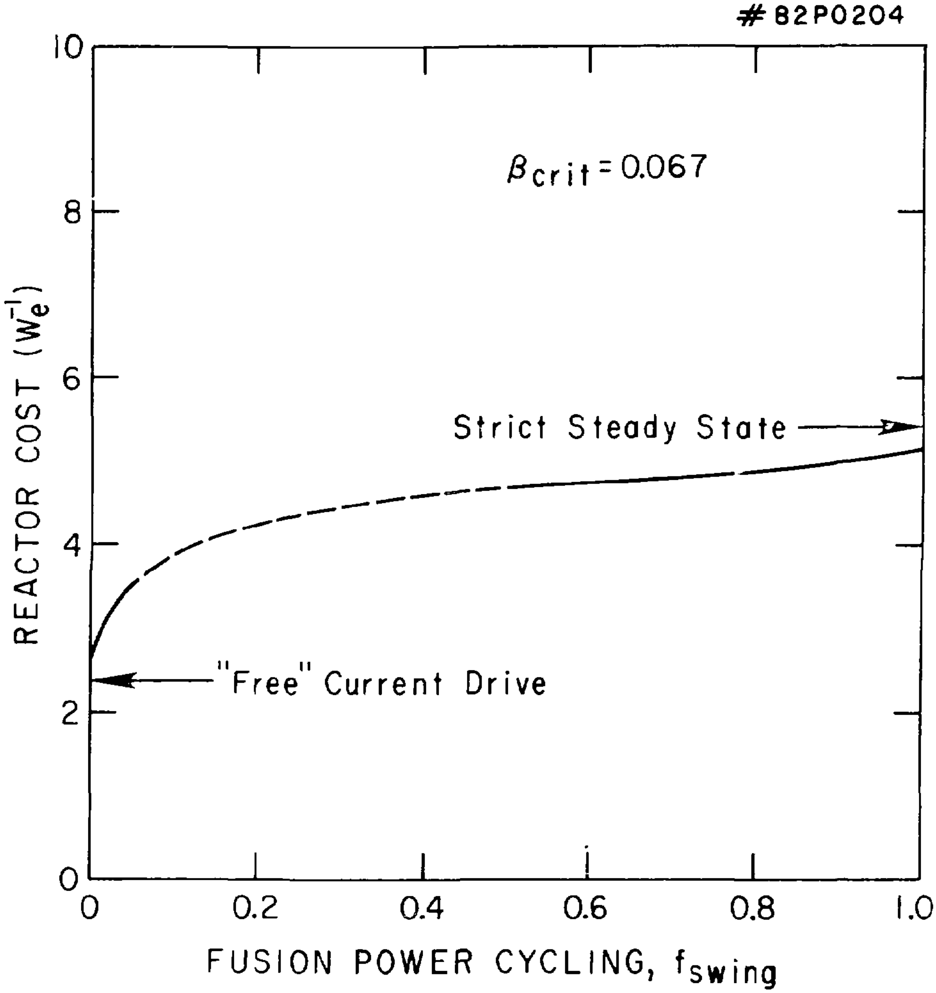



Fig. 2



#82P0204
$\beta_{crit}$ = 0.067
Strict Steady State
"Free" Current Drive
REACTOR COST ($W_e^{-1}$)
FUSION POWER CYCLING, $f_{swing}$
0
2
4
6
8
10
0
0.2
0.4
0.6
0.8
1.0


Fig. 3

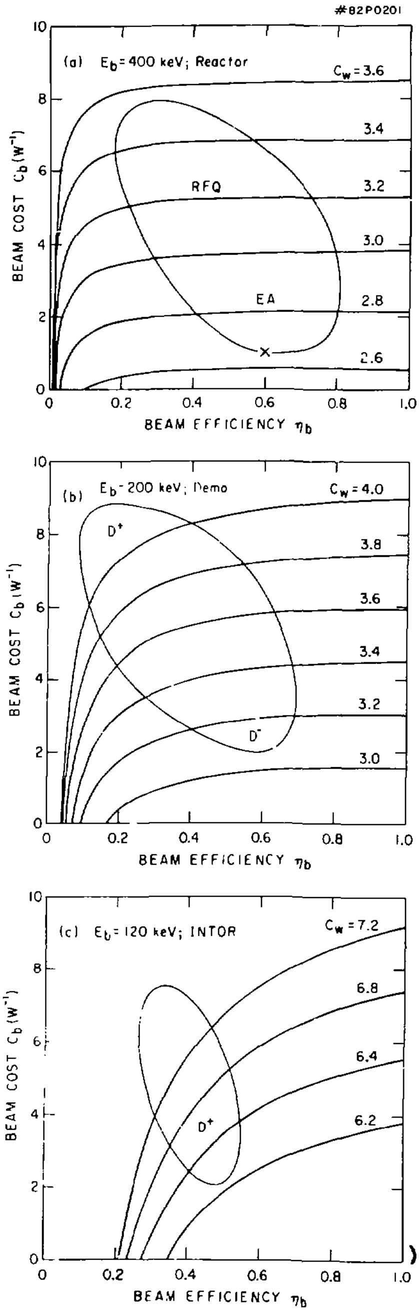

#82P0201
(a) $E_b$=400 keV; Reactor
$C_w$=3.6
3.4
3.2
3.0
2.8
2.6
RFQ
EA
BEAM COST $C_b$ ($W^{-1}$)
BEAM EFFICIENCY $\eta_b$
(b) $E_b$=200 keV; Demo
$C_w$=4.0
3.8
3.6
3.4
3.2
3.0
$D^+$
$D^-$
BEAM COST $C_b$ ($W^{-1}$)
BEAM EFFICIENCY $\eta_b$
(c) $E_b$=120 keV; INTOR
$C_w$=7.2
6.8
6.4
6.2
$D^+$
BEAM COST $C_b$ ($W^{-1}$)
BEAM EFFICIENCY $\eta_b$

Fig. 4