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LOW POWER HIGH WALL LOADING TOKAMAK REACTOR

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ABSTRACT

A summary of a recent design effort for a high wall loading, low power, compact tokamak reactor is presented. The design employs a bean-shaped plasma to reach the second stability regime, where high beta values are attainable. A point design was chosen based on a parameteric survey that was conducted using the Tokamak Systems Code (TSC). The most important parts of that survey are presented here. Investigation of different plasma confinement scaling laws shows that the chosen point design has a reasonable ignition margin.

INTRODUCTION

It is important, at this stage of development for fusion power, to examine the features that would bring such a limitless energy source to commercialization. In particular, one needs to decide what features of fusion reactors would allow rapid penetration into the marketplace. It is likely that the tokamak concept will be the first to achieve breakeven and, perhaps, the first to be the target of commercialization. In this paper, a design effort is described which investigated innovations and features that could help tokamak reactors achieve the most rapid penetration into a commercial market.

The design was guided by three major criteria. The first was to have a compact system; i.e. a system with high power density, high wall loading, and low mass utilization. The power density is defined[1] as the ratio of the total thermal power to the volume enclosed by and including the magnetic coils, and the mass utilization factor is defined as the weight of the first wall, drained blanket, shield, and coil divided by the total thermal power. For the same fusion power, a compact system is significantly smaller than a conventional one. With the right choice of materials, which will be under higher stresses in a compact system, a more economical reactor should result.

The second criterion of the design was to aim at a small system in both physical size and output power (~ 1000 MWt). Although "economy of scale" does not favor small size systems, they have the advantage of being affordable because of relatively small capital cost, and their shorter construction time. Consequently, small reactors should provide the fastest and safest (from an economic point of view) path to the realization of economical fusion power. Economic and technological issues of compact and small systems have been discussed in detail in recent review articles.[2] It is important at this point to examine the differences between a compact-small system and a conventional one. Table I lists some key parameters for the STARFIRE design[3] and the RIGGATRON design.[1] It is evident from the parameter comparison of the conventional design (STARFIRE) and the extremely compact design (RIGGATRON) that there is a wide range in the tokamak design space between these two design points.

Finally, a third criterion of the design was to make use of a bean shaped plasma to reach the high beta regime. High beta and high magnetic field are the major driving parameters for

Table I. Conventional and Compact Design Parameter Comparison

	STARFIRE	RIGGATRON
β (%)	6.7	20.
Plasma Radius (m)	1.94	0.34
Major Radius (m)	7.00	0.85
Aspect Ratio	3.6	2.5
Max. Field at TF Coil (T)	11.1	10-16[a]
Thermal Power (MW_{th})	4033.	1325.
Net Power (MW_e)	1200.	355.
Plasma Power Density (MW/m^3)	4.5	500.
Neutron Wall Loading (MW/m^2)	3.6	68.4
System Power Density (MW_{th}/m^3)	0.3	5.2 (LWR[b] ~ 20)
Mass Utilization ($tonne/MW_{th}$)	4.0	0.28 (LWR ~ .2)

a - normal magnet b - Light Water Reactor compactness. Since the magnetic field is limited by the magnet technology, the bean shaped plasma allows a major step in achieving compact tokamaks. Plasma indentation is controlled by a normal-conducting "bean coil" interlocking the superconducting toroidal field (TF) coils. We envision a scenario in which the bean shaped plasma is employed temporarily as an intermediate stage to reach the high beta regime. The plasma starts with a low beta, D-shape. The indentation and the beta of the plasma are then increased to reach a moderate beta bean shaped plasma. The beta is raised to reach the second stability regime for a D-shaped plasma; the indentation is then decreased to zero. With this scheme, the bean coil would be energized only during startup.

Previous work related to bean shaped plasmas has concentrated on MHD equilibrium and stability from a theoretical perspective. In this work we addressed reactor issues which arise when employing the bean shaped plasma. As it turned out, neither the interlocking of the TF coils by the bean coil, nor the power consumption in the bean coil (based on the above scenario) are major engineering problems for tokamaks employing bean shaped plasmas. The main difficulty was the placement of the poloidal field (PF) coils. Based on preliminary results of an MHD code,[4] the PF coils require large currents, if they are placed outside the TF coils, to achieve equilibrium. However, this code was not designed specifically to investigate bean shaped plasmas, and more work in this area is required.

PARAMETER SURVEY

To reach a point design that is in accord with the above criteria, an extensive parameter survey has been carried out with the aid of a modified version of the Fusion Engineering Design Center (FEDC) tokamak systems code (TSC),[5] assuming that the scaling laws in this code could be used for the bean shaped plasma. This code serves as a good tool for parametric surveys but, since most of its modules are based on simplified models of the relevant theories, its output should be checked with more sophisticated codes.

As mentioned earlier, there is a wide range in the tokamak design space that has not been examined before, especially for high power density designs. In this survey we studied the effect of some important parameters on the design space and showed how the fusion power as a function of the aspect ratio changes for different values of plasma radius, maximum field at the TF magnet, beta, wall loading, inboard shield thickness, etc. Some general assumptions and conditions for the design were established at the beginning of the study. These assumptions and conditions were:

A. <u>Steady State - RF Startup - RF Current Drive</u>
The recent experimental success[6] in using lower hybrid (LH) and ICRF (Ion Cyclotron Radio Frequency) waves in starting and driving the plasma current is very encouraging, and we assume that steady-state operation of tokamak reactors using RF is attainable. The use of RF heating and current drive has the advantage of eliminating the need of using ohmic heating, and thus removes an important constraint on the size of the inner core of the reactor; this is an important step toward a small compact tokamak. On the other hand, since the required RF power is proportional to the plasma volume, small reactors could be ignited with only a few megawatts of RF power.[7] Hence the use of RF in a small compact system is favorable.

B. <u>No Inboard Breeding</u>
For a compact, small reactor, inboard breeding has two unwanted effects. First it increases the total thickness of the inboard blanket/shield, which means an increase in the size of the reactor as well as an increase in the maximum field at the TF magnet. Second, it complicates the design and the maintenance of the reactor. With careful design of the outboard blanket, we found that sufficient tritium breeding could be achieved without the need for an inboard breeding blanket.

C. <u>Fusion Power</u>
As stated above, we aimed at a fusion power of about 1000 MW. However, for completeness, for the sake of comparison with other large, compact designs, and to show how the fusion power changes when one or more parameters change, we cover a wide range of the fusion power.

D. <u>Neutron Wall Loading</u>
A peak value of the neutron wall loading of ~ 10 MW/m^2 at the inboard wall is assumed. An optimized inboard shield thickness that can protect the magnet under this condition was found to be 66 cm. The inboard dimensions from the plasma edge to the TF winding pack are given in Table II.[8]

E. <u>Fixed Plasma Input Parameters</u>
Table III shows the plasma parameters that were assumed fixed in this survey. The choice of these parameters was based on the results and conclusions of previous conceptual designs.

In the following, we present some of the results of the parameter survey and show, in particular, the effect of beta, neutron wall loading, and inboard shield thickness on the design space. With respect to neutron wall loading, the independent parameter is the field at the plasma center. We modified the physics module in TSC so that it iterates on either the

Table II. Inboard Dimensions (cm)

Scrapeoff	6
First Wall	1
Shield	66
Gap	3
Cryostat	17
Total	93

Table III. Fixed Plasma Input Parameters

Elongation	1.6
Triangularity	0.3
Safety Factor	2
T_i (keV)	13
T_e (keV)	13
Z_{eff}	1.5
Z_{imp}	8

field at the plasma center or the aspect ratio (in case the maximum field at the TF magnet is specified) to achieve the required input value of the wall loading. The wall loading in this code is calculated as the ratio of neutron fusion power to the plasma surface area. The actual poloidal distribution of the neutron wall loading was found using the NEWLIT code.[8]

Figures 1 and 2 show the design space for an average neutron wall loading (Γ_{av}) of 10 MW/m^2 and for beta values of 10% and 20% respectively and for different plasma radii. Since the fusion power is proportional to $\beta^2 B^4$, it is clear from these figures that for the same plasma radius and power level, an increase of beta would decrease the required field at the plasma, the maximum field at the TF coil (B_{max}), and the aspect ratio. Decreasing B_{max} would make it possible to use NbTi magnets. The general effect of beta is to move the design space toward smaller size and lower power, in accordance with the second criterion of the design mentioned above.

Figure 3 shows the design space for a beta value of 20% and for 20 MW/m^2 average wall loading. The general effect of the wall loading (for the same value of beta) is to move the design space towards higher power, larger size, and higher field, as can be seen by comparing Fig. 2 and Fig. 3. It is evident that the first two general criteria of the design, i.e compactness (higher wall loading) and smallness (of size and power) are not parallel. To have a small reactor with very high wall loading and the same beta, a very high magnetic field (> 16 T) is required.

The inboard shield thickness d_s should be optimized to fully protect the magnet with the least possible thickness. This thickness determines the maximum field of the TF magnet and/or the field at the plasma center. Consequently,

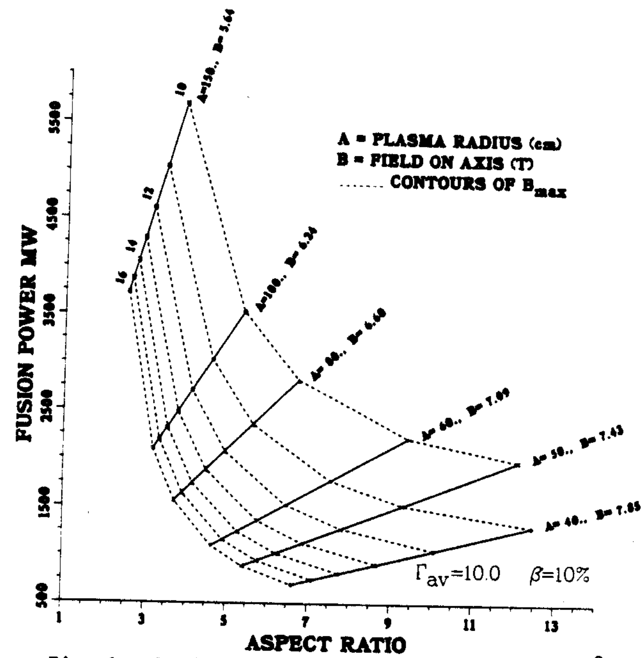

Fig. 1. Design space for β=10%, Γ_{av} = 10 MW/m^2.

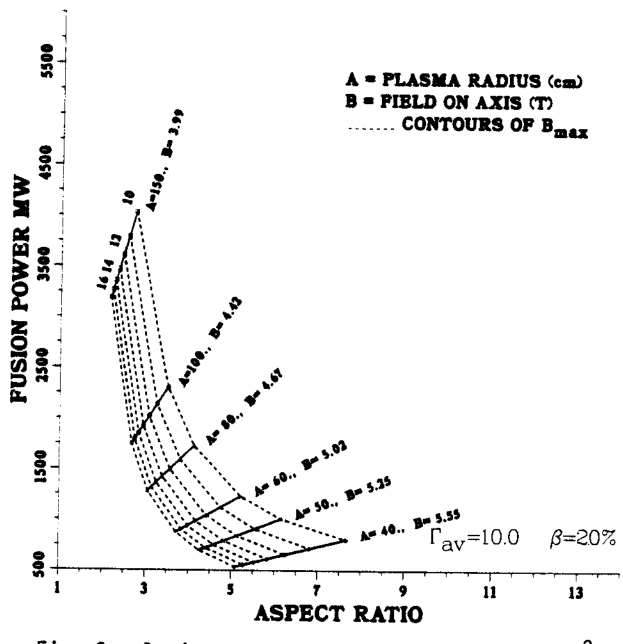

Fig. 2. Design space for β=20%, Γ_{av} = 10 MW/m^2.

changing d_s would affect the design space. To show this effect, we considered two values of d_s. Figure 4 shows part of the design space for Γ_{av} = 10 MW/m^2, beta = 20%, and d_s = 66 cm. The point design of the base case is shown by a

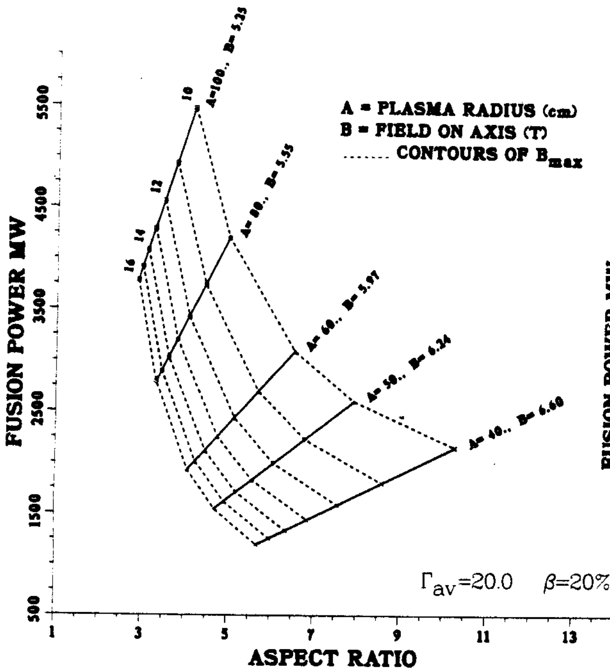

Fig. 3. Design space for β=20%, Γ_{av} = 20 MW/m^2.

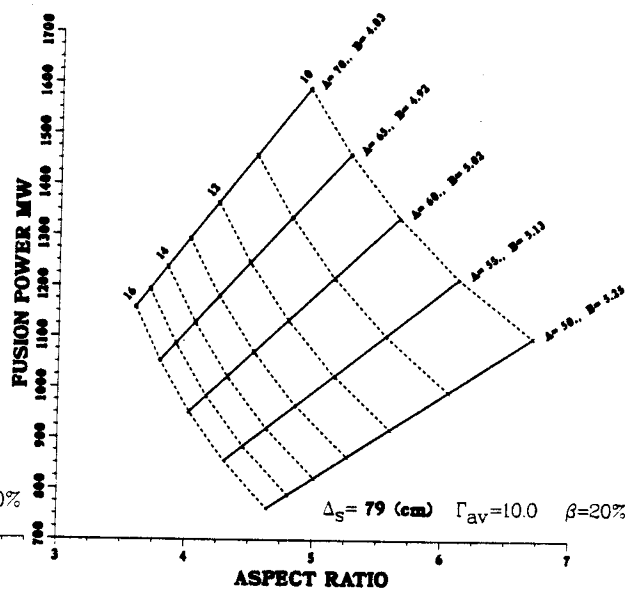

Fig. 5. Design space for Δ_s = 79 cm.

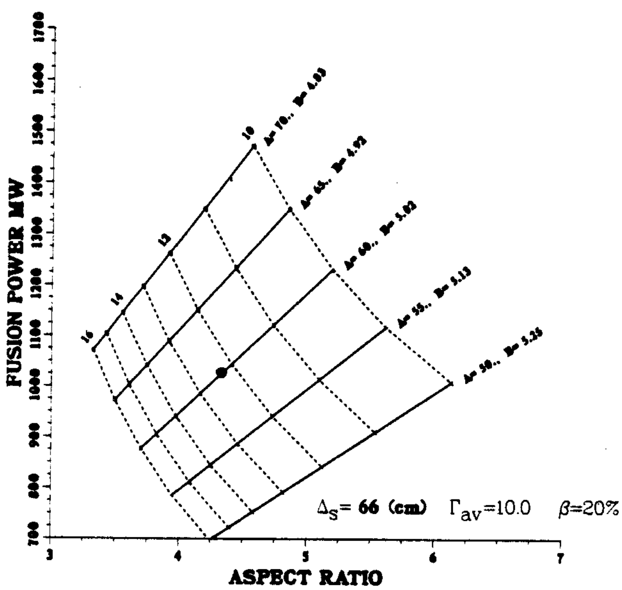

Fig. 4. Design space for Δ_s = 66 cm.

circle. Figure 5 shows the case of d_s = 79 cm for the same beta and Γ_{av}. To keep the same power, the same plasma radius, and the same aspect ratio, the maximum field will increase from 12.29 T (d_s = 66 cm) to 14 T (d_s = 79 cm).

On the other hand, to keep the same maximum field (12.29 T) and the same plasma radius, the aspect ratio increases from 4.33 (d_s = 66 cm) to 4.9 (d_s = 79 cm), and the fusion power increases by about 100 MW.

Along with the neutronics calculations,[8] this survey led to a reference design case with the parameters listed in Table IV. The plasma radius of 60 cm was selected to achieve adequate confinement. A lower value of the plasma radius would result in poorer confinement, although low B_{max} and small power could be achieved. On the other hand, for a larger value of the plasma radius, the fusion power and/or B_{max} would be outside the design boundaries. The beta value was chosen according to the scenario discussed above. The average neutron wall loading on the plasma surface is 10 MW/m^2, and the peak value of the inboard wall loading is 9.3 MW/m^2. The system power density of this design is 2.15 MW/m^3, which is about 7.2 times that of STARFIRE.

PLASMA CONFINEMENT SCALING

The parametric calculations described above were used to scope out a design point which turned out to have a rather small plasma; the minor radius is about 60 cm. An important consideration in such small plasmas is whether they will ignite. Of course, energy loss processes are not well known, especially in a reactor grade plasma. Various scaling laws have been developed for current tokamak experiments. In this section, we apply these scaling laws to see

Table IV. Reactor Parameters (Base Case)

Plasma Radius (cm)	60.
Major Radius (cm)	260.
Inboard Shield Thickness (cm)	66.
B_{max} (T)	12.29
B_o (T)	5.02
Fusion Power (MW)	1025.
Inboard Peak Neutron Load (MW/m^2)	9.3
Max. Neutron Load (MW/m^2)	11.76
System Power Density (MW/m^3)	2.15
β	0.20
β_p	3.3
n_i (m^{-3})	4.2×10^{20}
n_e (m^{-3})	4.2×10^{20}
τ_e (s)	0.395

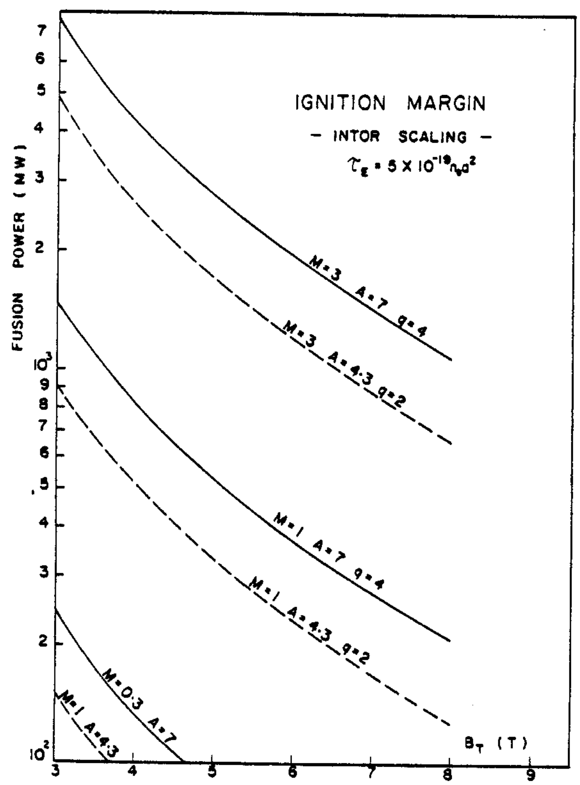

Fig. 6. Ignition versus fusion power and B_T for INTOR scaling.

if these small reactor plasmas will ignite. We recognize that this represents a large extrapolation from the present database, but no reasonable alternative is available at this time.

The scaling laws investigated are INTOR,[9] Ohmic,[10] Mirnov,[11] L-mode,[10] H-mode, and a modified H-mode scaling based on ASDEX[12] data. The following parameters are fixed in the scaling study: beta (= 20%), average ion and electron temperature (= 13 keV), aspect ratio, A, and MHD safety factor, q. For A and q we consider 2 cases: (1) A = 4.3, q = 2, which is the chosen design point, and (2) A = 7, q = 4, which corresponds to a higher fusion power case (1500 MW) but may ease problems with the poloidal coil set required for MHD equilibrium. The ignition parameter, M, used in this study is defined as the ratio of the alpha particle heating power to the power lost by transport across the magnetic field. For D-T fusion at a temperature of 13 keV, M is given by

$$M \equiv \frac{n\tau_E \ (s/cm^3)}{2 \times 10^{14}} .$$

Here, n is the average density and τ_E is the global energy confinement time. With this definition of M, M = 1 is an ignited plasma in energy balance. M > 1 is super-ignited and the plasma temperature will rise. Alternatively, one can view M > 1 as providing a safety factor for errors in the scaling law, or against additional power losses not considered in the scaling law.

The procedure is first to select the on-axis toroidal magnetic field, B_T. The plasma density, size, and fusion power are then determined. The scaling law then determines the ignition parameter. The results are presented as plots of the fusion power for given M, A, and q, as a function of B_T. In this way a large set of possible design points is considered together.

Figure 6 shows the results with INTOR scaling. Our design point (B_T = 5 T, A = 4.3, q = 2) ignites at about 600 MW and has an ignition margin of about 2 at the design power level. The higher aspect ratio case requires a higher fusion power for the same M. Ohmic scaling, which describes a large variety of tokamaks in the ohmic heating phase, is very favorable for ignition and even rather small plasmas will ignite according to this scaling law. Mirnov scaling has been used in the TFCX studies. This scaling favors lower aspect ratio and q, in comparison to previous scalings. In this case, the 1000 MW design point (A = 4.3) has an ignition margin of about 2.5, but the higher aspect ratio case does not ignite.

Confinement studies of tokamaks with strong neutral beam heating have shown more pessimistic results for the energy confinement time. There are two modes of operation, the L-mode (low confinement) and the H-mode (high confinement). For L-mode confinement, τ_E degrades with increasing injection power. How does one apply this to a reactor? If the injection power is replaced by the alpha heating power in the L-mode scaling law, then the results are very pessimistic and anything near ignition is not

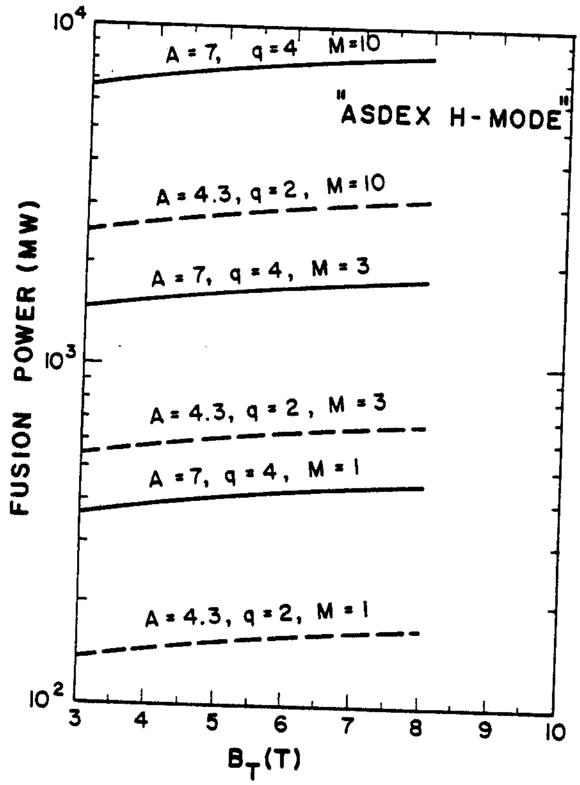

Fig. 7. Ignition parameter space for modified H-mode scaling.

possible. In the H-mode, an improvement in τ_E of about a factor of two is usually seen. If this factor of two is used for the scaling of a reactor grade plasma, then ignition is still not possible. The H-mode scaling law is obtained from a large number of tokamaks, and the data set has a lot of scatter. The ASDEX experiment does not see the continued degradation of confinement with increasing injection power when the plasma is in the H-mode. This suggests that using the L-mode scaling law with a factor of 2 for improvement due to the H-mode may be unduly pessimistic. A modified scaling law with τ_E independent of injection power can be obtained. This is

$$\tau_E = 8.6 \times 10^{-11}\ I_p R^{1.75} a^{-0.37} \kappa^{0.5}$$

where I_p (amp) is the plasma current, R (cm) the major radius, a (cm) the minor radius, κ the ellipticity, and the coefficient has been obtained by fitting to ASDEX data. The resulting ignition margin for this scaling is shown in Fig. 7. In this case, one gets a reasonable ignition margin.[11]

From these considerations, the 1000 MW design point has a reasonable ignition margin even though the plasma is very small (60 cm). An inadequate ignition margin is obtained only for L-mode scaling. The modified H-mode scaling based on ASDEX data is more favorable and leads to a good ignition margin.

CONCLUSIONS

A parametric survey leading to a small, compact, tokamak reactor point design has been presented. We have found that such designs are possible if the second stability regime is utilized. The resulting plasmas are small, but present scaling laws generally indicate that the plasma will be ignited with a reasonable margin. The principal problem remaining is the poloidal magnet set; suitable coil locations consistent with both good MHD equilibrium and reactor engineering constraints are difficult to find. This problem requires further investigation.

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