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COMPACT IGNITION TEST REACTOR (CITR)

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Abstract

Design considerations have been developed for a compact ignition test reactor (*CITR*). The objectives of this tokamak device are to achieve ignition; to study the characteristics of plasmas which are self heated by alpha particles; and to investigate burn control. To achieve a compact design the toroidal field magnet consists of copper-stainless steel plates to accommodate relatively high stresses; it is inertially cooled by liquid nitrogen. No neutron shielding is provided between the plasma and the toroidal field magnet. The flat top of the toroidal field magnet is ~10 s. Strong auxiliary heating is employed. In one design option, adiabatic compression in major radius is employed to reduce the neutral beam energy required for adequate penetration; this *compression boosted* design option has a horizontally elongated vacuum chamber; illustrative parameters are a compressed plasma with $a = 0.50$ m, $R = 1.35$ m, $B_T = 9.1$ T and a neutral beam power of $\sim$ 15 MW of 160 keV D^o beams. A design option has also been developed for a *large bore* device which utilizes a circular vacuum chamber. The *large bore* design provides increased margin and flexibility; both direct heating with *RF* or neutral beam injection and compression boosted startup are possible. The large bore design also facilitates the investigation of high-Q driven operation. Illustrative plasma parameters for full use of the large bore are $a = 0.85$ m, $R = 1.90$ m and $B_T = 7.5$ T.

I. Introduction

In a deuterium tritium plasma, self heating can be provided by alpha particles produced in the D+T $\rightarrow \alpha(3.5\ \text{MeV}) + n(14.1\ \text{MeV})$ reaction. At ignition the alpha power is sufficient to balance the plasma losses; in this case $Q \rightarrow \infty$, where Q = Fusion power/Auxiliary Heating power. The high values of Q needed for many fusion reactor appplications are obtained by operation at or near ignition.

In this paper design options are developed for a compact tokamak ignition test reactor (*CITR*). The *CITR* utilizes high performance copper magnets which produce moderately high magnetic fields (7 T-10 T on axis). The main objectives of the *CITR* are:

- to study the characteristics of self heated plasmas
- to demonstrate ignition
- to investigate stability control over an extended period of time ($\gtrsim$ 5 s).

In addition, the *CITR* might also be used

- to study methods to vary fusion power production at ignited thermal equilibrium
- to investigate refueling, impurity generation and helium buildup (pulse lengths of ~ 60 s can be obtained for unignited plasmas with magnetic fields ~ 4T - 5T).

The use of copper coils removes the need for the neutron shielding that would be required if superconducting magnets were employed: furthermore, peak fields can be higher. The machine size and cost can thus be substantially reduced relative to devices which employ superconducting magnets.

The toroidal field (*TF*) magnet utilizes a Bitter plate design consisting of interleaved copper and steel plates, similar to the design employed in the *ALCATOR* devices [1]. A large number of these plates are stacked side by side, separated by planar insulation. The magnet is inertially cooled. Precooling to liquid nitrogen temperatures allows for relatively long pulse lengths: the flat top of the *TF* magnet is $\sim$ 10 s in the presence of neutron heating. Access is provided for strong auxiliary heating by neutral beam injection or *RF* heating.

The design characteristics have evolved from previous ignition test reactor design studies [2,3,4].

II. Features of Ignited Operation

In order to project the plasma parameters required for ignition, it is assumed that the electron energy confinement time is determined by the *ALCATOR* empirical scaling law [5]

$$(\tau_e)_{emp} = 0.5(\tau_E)_{emp} = 3.8 \times 10^{-21} n\, a^2 \tag{1}$$

where n is the average electron density in m^{-3}, a is the minor radius in m and τ_e is in s. The factor of 0.5 in (1) is due to the fact that $(\tau_E)_{emp}$ is defined in reference 5 as the global energy confinement time, even though most of the heat flux is due to the electrons. It is assumed that the ion energy confinement time, τ_i is given by neoclassical theory [6]. In this case τ_i is generally much greater than τ_e. The *ALCATOR*-empirical scaling law implies that the minor radius required for ignition scales as

$$a_{ign} \sim \frac{1}{n}\sqrt{(n\tau_e)_{ign}}, \tag{2}$$

where $(n\tau_e)_{ign}$ is the value of $n\tau_e$ required for ignition at a given temperature [7].

β_T, the ratio of the plasma pressure to the magnetic field pressure, is defined by

$$\beta_T = \frac{\langle nkT_i + nkT_e + \overline{E_{fp}n_{fp}}\rangle}{B_T^2/2\mu_o}$$

n, T_i, T_e and B_T are the spatially varying plasma density, ion temperature, electron temperature and toroidal magnetic field, respectively. $\overline{E_{fp}n_{fp}}$ represents the contribution of the fast fusion products to the plasma pressure. The angle brackets in (2) represent a radial average. Using (1) and (2), requirement for ignition can be expressed in terms of a requirement on $\beta_T a B_T^2$:

$$\beta_T a B_T^2 \sim (na)\frac{\langle nkT_e + nkT_i + \overline{E_{fp}n_{fp}}\rangle}{n} \sim \sqrt{(n\tau_e)}\frac{\langle nkT_e + nkT_i + \overline{E_{fp}n_{fp}}\rangle}{n} \tag{3}$$

From (3), the parameter $\beta_T a B_T^2$ at ignition, $(\beta_T a B_T^2)_{ign}$, depends only on the general ignition requirement on $n\tau_e$, T_e, T_i, τ_i/τ_e and on the plasma profiles [7]. This parameter is therefore independent of specific machine parameters.

The temperature dependence of $\beta_T a B_T^2$ at ignition is shown in Figure 1. It is assumed that there are no impurities, that $\tau_i \gg \tau_e$ and that the density and temperature profiles are parabolic. At any point above the ignition curve, the heating input from the fusion process is larger than the power loss. It can be seen that the value of $\beta_T a B_T^2$ required for ignition is minimum at central ion temperatures $T_{io} \sim 15$ keV.

Because of uncertainty in scaling it is desirable to provide margin. A margin of ignition in confinement is defined as

$$MI = \frac{(n\tau_e)_{emp}}{(n\tau_e)_{ign}} \sim \frac{\beta_T^2 a^2 B_T^4}{(\beta_T^2 a^2 B_T^4)_{ign}} \tag{4}$$

Here $(n\tau_e)_{emp}$ is the value of the confinement parameter predicted by the empirical scaling law. The largest value of β_T is determined by MHD instabilities. It is assumed that for a circular plasma, this value of β_T is given by $\beta_T = \frac{0.10}{A}$, where A is the aspect ratio. The scaling of β_T with $1/A$ is consistent with MHD calculations [8] while $\beta_T \sim 3\%$ at $A \sim 3.4$ is consistent with recent results in ISX-B [9]. The margin of ignition provides safety against the presence of impurities, a higher value of thermal conduction loss and uncertainities in the achievable value of β_T.

If the energy loss mechanism is given by the $ALCATOR$ empirical scaling law for the electrons and neoclassical ion energy transport, an ignited plasma is unstable for central temperatures up to $\sim$ 45 keV [7]. The runaway time at central temperatures $\sim$ 20 keV is comparable to the electron energy confinement time. The runaway time is increased (and the temperature at which thermal stability is attained is reduced) if radial motion of the plasma is allowed [10,11].

The $CITR$ will be used to study thermal stability control. Thermal stability control is required in an ignited plasma to permit steady operation at whatever temperatures optimize plasma performance. Optimization of plasma performance includes minimization of β_T required for ignition, maximization of fusion power density and minimization of adverse effects from plasma-wall interactions. Passive thermal stability might be provided by ripple induced ion energy transport loss which increases rapidly with increasing temperatures [12]. If radial motion of the plasma is allowed, a very small amount of ripple induced ion energy transport ($<$ 10% of the total power loss) could provide thermal stability over a wide temperature range ($T_{io} >$ 15 kev) [13]. If automatic thermal stability is not obtained, then some external means for thermal control must be supplied. Schemes which have been proposed for active thermal stability control include adiabatic compression and decompression [10] and high Q operation with variable heating [14].

The $CITR$ might also be used to study methods to vary fusion power production. Fusion power production from an ignited tokamak can be varied by use of impurities or toroidal field ripple to increase the plasma power loss; the fusion power level at thermal equilibrium is then adjusted so that the alpha power balances the plasma power loss [12,15,16].

III. Compression Boosted Design Option

Adiabatic compression in major radius [17] can be used to increase neutral beam penetration [18]. It is desirable that the degree of beam penetration be similar to that in present successful neutral beam heating experiments. Edge heating (heating deposition profile with an off axis peak) may result in increased impurity buildup: higher temperatures near the edge of the plasma and high energy charge exchange neutrals may result in increased impurity generation. Furthermore, edge heating is inefficient and leads to increased beam power requirements.

In a compression-boosted startup na and $n\tau_e$ are increased by compression after neutral beam heating of the initial plasma. According to the *ALCATOR*-empirical scaling law for the energy confinement (1), the value of $n\tau_e$ for the final plasma is related to the value of $n\tau_e$ of the initial plasma by

$$(n\tau_e)_f = C^3(n\tau_e)_i \tag{5}$$

where C is the compression ratio in major radius. The subscripts i and f refer respectively to the initial precompressed state and to the final compressed state.

The effect of the compression ratio on the neutral beam energy required to provide centrally peaked beam deposition profiles at ignition is shown in Figure 2 for a central temperature at ignition of 15 keV. Parabolic density profiles are assumed. The upper curve is for beam deposition profiles that are strongly peaked in the center, while the lower curve is for moderately peaked deposition profiles. Near perpendicular injection of a monoenergetic beam is assumed. For neutral beam energies ~ 160 keV, a compression ratio of $\sim$ 1.5 provides moderately peaked deposition profiles.

Compression is also useful in assuring the attainment of high temperatures even if the energy confinement time decreases with temperature.

Illustrative machine parameters for a *compression boosted* design option are shown in Table I. Figure 3 shows an elevation view. The vacuum chamber is horizontally elongated to accomodate the requirements for adiabatic compression with minimum increases in the size and stored energy of the *TF* magnet. There are large horizontal ports required for neutral beam injection and small vertical ports for diagnostics. The parameters of the *TF* coil are shown in Table II.

The *TF* magnet is initially cooled to liquid nitrogen temperature (77 K) and is inertially cooled during the pulse. Cooling the magnet down to liquid nitrogen temperature after the end of a 9 s flat-top pulse (with 5 s of ignited operation) requires 10,000 liters of liquid nitrogen. The pulse length is limited by heating of the TF coil from both ohmic dissipation and neutron irradiation. There is a tradeoff between the pulse length and the limit on the magnet lifetime due to cycling fatigue; as the pulse length increases the peak temperature in the TF coil increases, reducing the lifetime. The temperature of the copper in the throat of the TF coil for 9 s flat-top pulses (including 5 s of ignited operation) is 270 K; in this case the copper fatigue life is $> 3,000$ cycles (allowing for a safety factor of 10 in the fatigue data). The temperature limitations set by fatigue can be surpassed for a few pulses without reducing significantly the lifetime due to cycling failure. The lifetime of the magnet can also be limited by neutron damage of the insulation; however it may be possible to obtain $50,000$ burn-seconds with organic insulation [19].

The flat top time could be significantly increased by operation at reduced fields, as $\tau_{flat\ top} \sim \frac{1}{B_T^2}$. Unignited operation at $B_T \sim 5$ T could result in pulses that are ~ 60 s long. Long pulse operation at reduced fields does not affect the fatigue lifetime of the magnet. This opens interesting possibilities for the study of long-pulse operation of unignited plasmas.

The compression takes place in ~ 150 msec; it is estimated that the energy stored in the vertical field will increase during compresion by ~ 50 MJ (resulting in a peak power of 400 MW). Inductive energy storage elements are being considered to provide the energy required during the compression [3].

The neutral beam power required is ~ 15 MW of 160 D^o beams with $\gtrsim$ 85% D^+ in the source. The heating pulse length is ~ 1 sec. Access is provided for ~ 25 MW of neutral beam heating power. Six ports, each with ~ 0.2 m^2 free area are utilized; two additional ports are used for pumping. To prevent the escape of fast injected ions, the magnetic field ripple must be kept small. The plasma current of the initial plasma is 2.5 MA and $A \sim 3$. This amount of current should provide good confinement of the injected 160 keV ions [20].

Details about the magnetics design have been described elsewhere [3,21,22]. Design solutions have been developed for key problems, such as the minimization of the ripple from the large neutral beam ports and from the flanges; torsion from the interaction of the vertical and toroidal fields; and disassembly of the TF magnet. Both Bitter and tape-wound toroidal field magnet designs were developed for the $ZEPHYR$ device proposed for construction at the Max Planck Institut fur Plasmaphysik at Garching [4].

IV. Large Bore Option

The flexibility and margin of ignition of the *CITR* can be increased by changing the horizontally elongated bore of the *compression boosted* design option into a circular bore. This *large bore* option provides the capability to run a much larger plasma with direct heating while maintaining the possibility of using compression boosted heating scenarios with smaller plasmas [21]. Direct heating of the large bore plasma would be either by *ICRH* or by neutral beam injection. The *large bore* option is similar to *PITR* [23] but is considerably more compact and uses magnets with higher stress and current density.

With the use of a larger minor radius less reliance in placed on very high density operation. The *large bore* option also facilitates the study of high-Q driven operation and the possibility of utilizing elongated plasmas.

Illustrative parameters for the *large bore* option are given in Table III. The stored energy in the *TF* magnet is 1.4 GJ and the weight of the magnet is 600 metric tons. For a full size circular plasma with a minor radius of 0.85 m and a magnetic field of 7.5 T, a margin of ignition of 3 can be obtained. Larger margins ($\gtrsim$ 4) might be obtained with an elongated plasma. If desired, a compression boosted scenario could be used to obtain a margin of ignition of 1 with a compression ratio of 1.4 .

The flat top of the *TF* coil is $\sim$ 15 seconds if the peak temperature of the magnet is limited to 270 K.

Table IV provides a comparison between the parameters of the large bore option and the parameters of *TFTR* [24] and *INTOR* [25].

V. Conclusions

The use of high performance copper magnets makes possible the design of a compact ignition test reactor. The *CITR* could play a complementary role to a fusion engineering device (*FED* [26]). A *FED* operated at low Q could be optimized for the study of fusion engineering and long pulse, high duty factor performance. Information on ignited operation and burn control would be provided by a *CITR* with considerable margin and flexibility in order to meet its goals.

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List of Figures

Table I

Illustrative Parameters for

Compression Boosted Design Option

R (m)		1.35
a (m)		0.5
B_T (T)		9.1
I_p (A) (q at limiter = 2.5)		3.4×10^6
Compression Ratio		1.5
Average tensile stress in copper in TF Magnet Throat (MPa)		250
Stored energy in TF magnet (GJ)		0.87
Weight of TF magnet (tonnes)		420
Beam Energy (KeV)		160
Beam Power (MW)		15
$MI = \frac{(n\tau_e)_{emp}}{(n\tau_e)_{ign}}$	1	1.5
β	3%	3.7%
T_{io} (KeV)	15	15
n (m^{-3})	3.0×10^{20}	3.7×10^{20}
Fusion power (MW)	50	80
Equivalent Neutron Wall Loading (MW/m^2)	0.9	1.4

Table II

TF Magnet Characteristics for
Compression Boosted Design Option

B_T (T)	9.1
Peak field (T)	16.6
Copper-stainless steel ratio in throat	2:1
Number of plates	256
Plate current (A)	2.4×10^5
Copper average Current density in throat (A/m^2)	7.5×10^7
Dimensions	
Height (m)	2.40
Outer radius (m)	3.30
Inner radius	0.4
Average stress in copper in magnet throat (MPa)	250
Weight (tonnes)	420
Stored energy (GJ)	0.87
Resistive power (after 9 s flat top)(MW)	< 100
Peak power of TF power supply (MW)	150
Peak magnet temperature (9 s flat top)(K)	270
Energy dissipation (9 s flat top)	
electrical (GJ)	0.8
nuclear (GJ)	0.3
Number of 9 sec. full power pulses (including margin > 10 in fatigue life)	3000

Table III

Illustrative Parameters for Large Bore Design
(Full size circular plasma)

R (m)	1.9
a (m)	0.85
B_T (T)	7.5
I_p (A)(q at limiter = 2.5)	5.8×10^6
Average tensile stress in copper in TF coil coil throat (MPa)	190
Stored energy in TF magnet (GJ)	1.4
Weight of TF Magnet (tonnes)	600
Auxiliary heating power (MI = 1)(MW)	16

$MI = \frac{(n\tau_e)_{emp}}{(n\tau_e)_{ign}}$	1	3
β	2.6%	4.5%
n (m^{-3})	1.8×10^{20}	3.1×10^{20}
T_{io} (KeV)	15	15
$(\tau_e)_{emp}$ (s)	0.5	0.9
Fusion Power (MW)	80	240
Neutron Wall loading (MW/m^2)	1.2	3.6

Table IV

Comparison of Illustrative Large Bore CITR Design
with TFTR and INTOR

	TFTR	CITR	INTOR
R (m)	2.5	1.9	5.0
a (m)	0.85	0.85	1.2
B_T (T)	5	7.5	5.0
Weight of TF magnet and shielding (tonnes)	1200	800	> 8000
Stored energy in TF magnet (GJ)	1.4	1.4	20
Flat-top pulse length (s)	1-2	~ 15	~ 60
β_T	3%	4.5%	6%
Plasma Shape	Circular	Circular	Elongated
$MI = \frac{(n\tau_e)_{emp}}{(n\tau_e)_{ign}}$ (at T_{io} = 15 KeV)	0.35	3	2.1

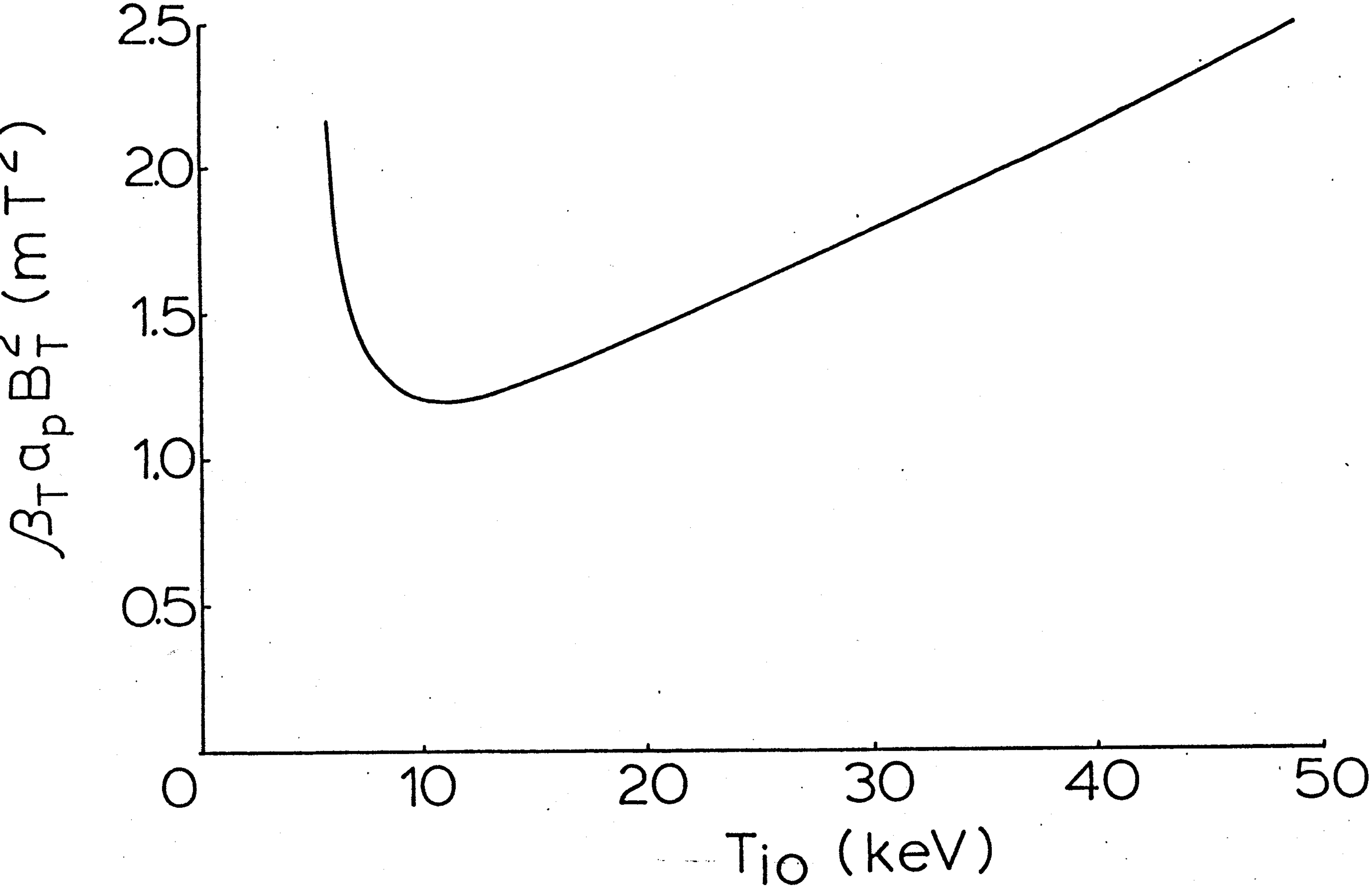

Figure 1

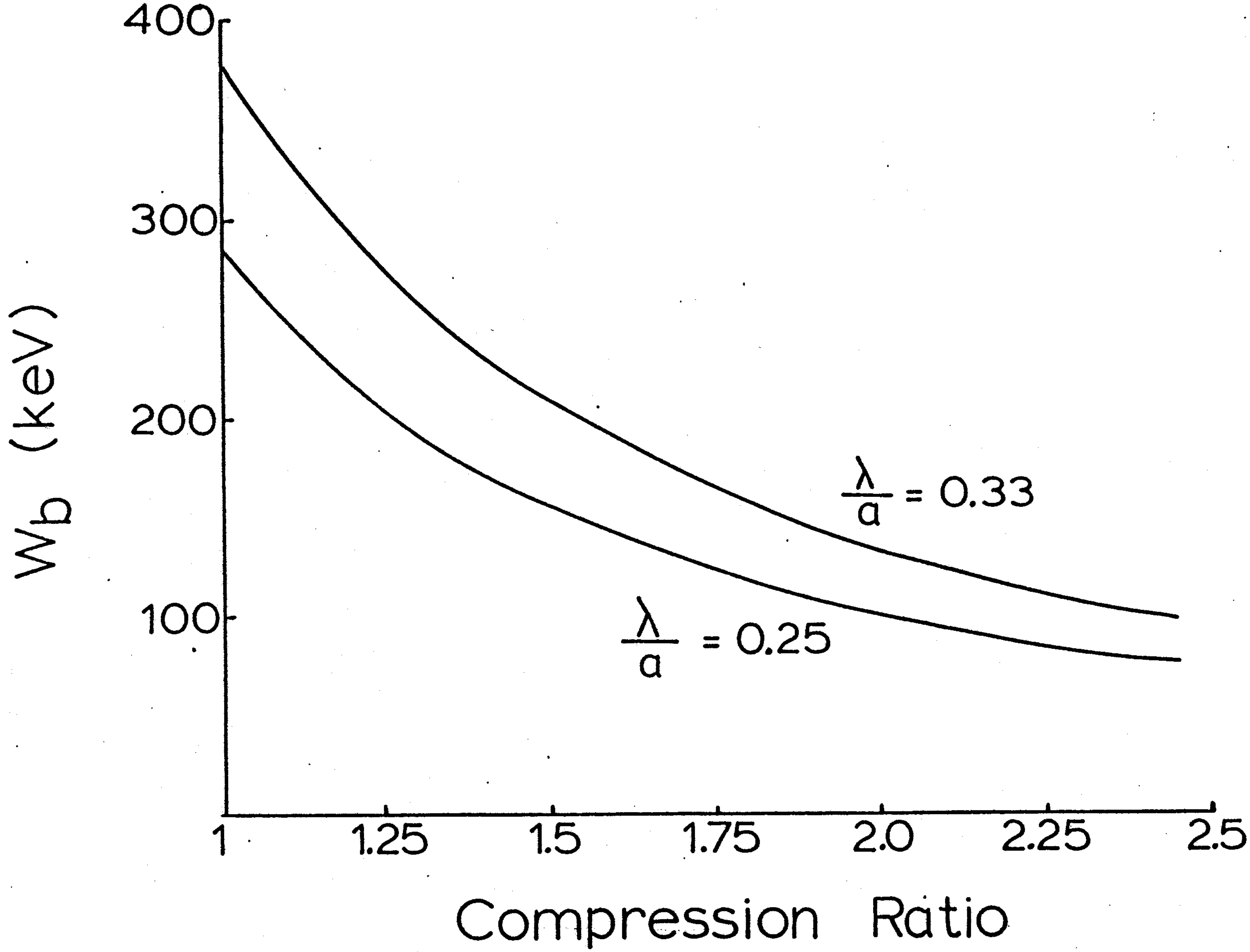
400
300
200
100
W_b (keV)
$\frac{\lambda}{a}$ = 0.33
$\frac{\lambda}{a}$ = 0.25
1
1.25
1.5
1.75
2.0
2.25
2.5
Compression Ratio

Figure 2

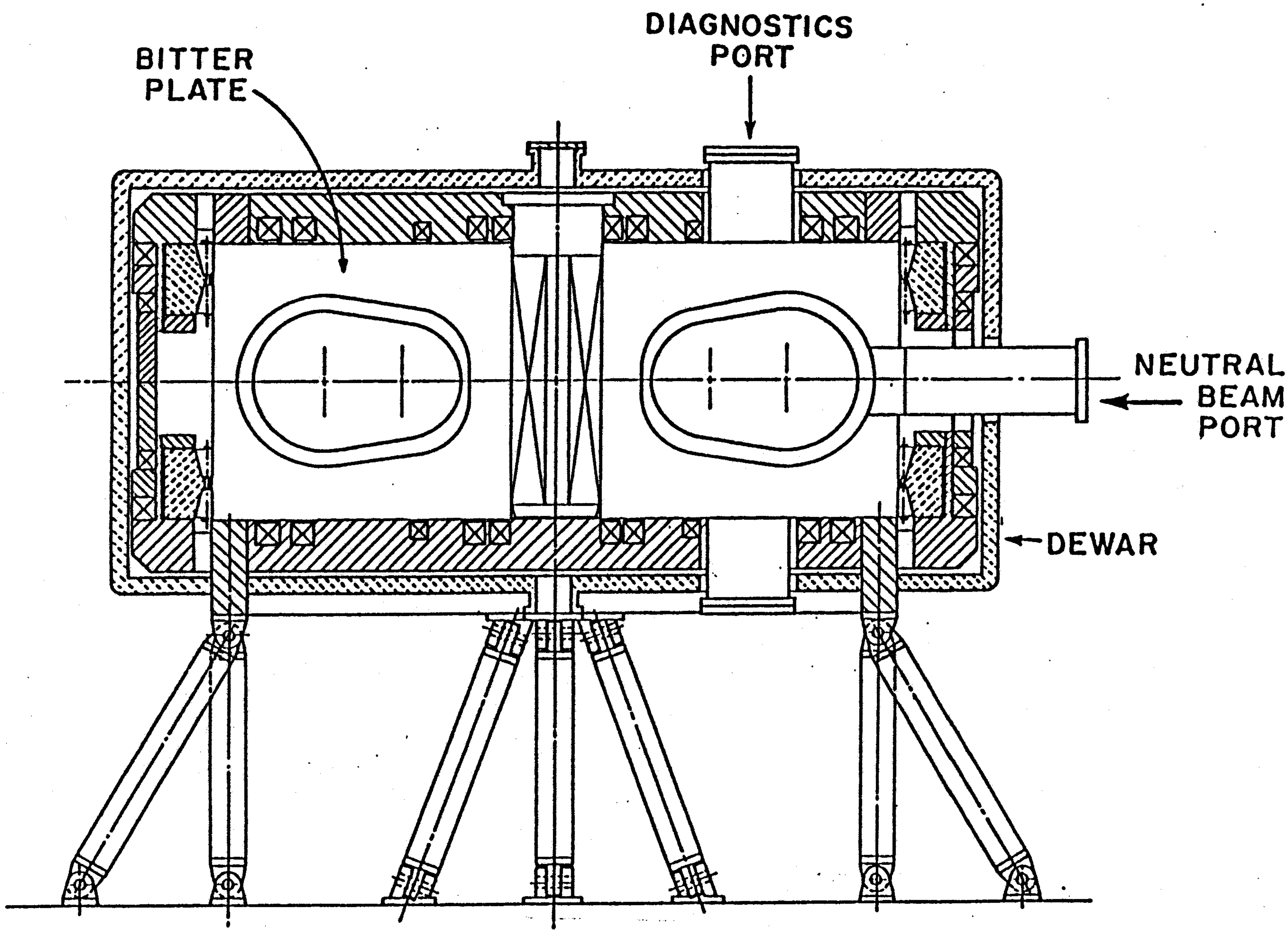

Figure 3