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INTOR SCOPING STUDY
—CONCEPTUAL DESIGN STUDY OF FY86 FER—

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The INTOR Scoping Study was performed by using the new-TORSAC which was developed from TORSAC (TOkamak Reactor Systems Analysis Code). The ASDEX type τ_E scaling was used in this scoping studies.

The difference of the possible burn time between quasi-steady state and inductive operation scenario decreases with increasing plasma elongation and triangularity. The maximum stored energy, total PF coil Ampere turn and the capital cost are less in single-null divertor case than in double-null divertor one. The effect of increasing toroidal field is not expected to be large. The capital cost has a weak minimum point at plasma elongation of 1.8 ~ 1.9 in the case of adopting radial access at reactor maintenance. The capital cost of inductive operation reactor is expected to be higher than quasi-steady state operation one by approximately 2 ~ 4 %. No clear merit can be expected by using ergodic limiter concept since the ergodic region increases machine size. The capital cost can be significantly reduced (10 ~ 20 %) by adopting steady state operation scenario, though the energy multiplication factor and wall loading would be low compared with other operation scenario.

Keywords: INTOR, new-TORSAC, Systems Analysis, Plasma Design, Relative Capital Cost, Possible Burn Time, Single-Null, Double-Null, Plasma Elongation, Operation Scenario, Energy Multiplication, Radial Access, Oblique Access

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INTORスコーピングスタディ
── 次期大型装置設計（FY86 FER）──

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（1987年7月8日受理）

システム解析コード TORSAC（TOkamak Reactor Systems Analysis Code）を発展させて作成した new－TORSAC を用いて，INTOR設計に関するスコーピングスタディーを行った。ASDEX型の τ_E スケーリング則を用いた。

準定常運転と完全誘導運転の間にある可能燃焼時間の差はプラズマの惰円度，三角度を増大させると減少する。PFコイルの最大蓄積エネルギー，全電流及び炉のコストは，ダブル・ヌル・ダイバータに較べシングル・ヌル・ダイバータの場合の方が小さい。TFコイルの最大磁場を増加させる事は炉寸法，コストに殆んど影響を及ぼさない。水平方向近接の分解修理法を採用した場合，炉コストの最小値はプラズマ惰円度1.8～1.9に存在するが浅い極小点である。完全誘導運転は準定常運転に較べ2～4％炉コストが大きくなる。エルゴディックリミタを採用すると炉寸法が増加し，この採用による利点が見つけ難い。定常運転を採用すると炉コストがかなり小さくなる（10～20％）が，エネルギ増倍率Q及び中性子壁負荷がかなり減少する。

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Contents

目　　　　次

1. General Description on Systems Code

The TORSAC (TOkamak Reactor System Analysis Code) systems code was revised for the scoping study of next generation tokamak devices. The new TORSAC consists basically of four parts, i.e. (1) plasma design, (2) structure design, (3) magnetic field calculation and (4) miscellaneous. First of all, plasma parameters are calculated so as to provide desired performance objectives (e.q. ignition margin, burn time, wall loading, fusion output power, etc.). Then, the radial and vertical builds of machine are defined, based on the calculated plasma parameters and other necessary input data(thickness of clearances, blanket, vacuum vessel). The configurations of the first wall, blanket, shield, vacuum vessel and toroidal field (TF) coils are automatically generated in the code. The optimal PF coil locations and their sizes are determined by plasma equilibrium calculations. The operation patterns of PF coil currents are obtained by superposing Ohmic heating (OH) flux component on plasma equilibrium component, according to the assigned ($T_e(t)$, $n_e(t)$, etc.) and calculated (OH flux consumption, $L_p(t)$, $I_p(t)$, etc.) scenario data. In determining PF coil current patterns, the magnetic fields in the PF coils are calculated and are limited to less than the assigned maximum field. This limitation on peak magnetic field in PF coils determines burn time and/or available OH flux change. Based on the PF coil current operation patterns, the key parameters on PF coil power supply are calculated (Max. Ampere x Max. V, stored energy, peak power, etc). Other parameters such as pumping requirements, cooling requirements and cost of reactor are also calculated.

In the following paragraphs, brief descriptions are provided for the main routines of this systems code [1].

1.1 Plasma design

Plasma parameters can be chosen to satisfy one of the following performance combinations in our systems code, i.e. (1) ignition margin and burn time/available Ohmic heating (OH) flux change, (2) ignition margin and location of TF coil inner leg, (3) ignition margin and neutron wall loading, (4) ignition margin and fusion power, and (5) neutron wall loading and fusion power. The simplified relation is used to provide the assigned burn time or available OH flux change in the step of plasma design. In this scoping study, the option (1) is used for inductive operation and for quasi steady state operation, and (2) for full non inductive current drive operation scenario, respectively.

The power balance in a plasma is calculated by a zero dimensional steady state model. Plasma ion temperature, T_i, and electron temperature, T_e, are assumed to be equal to each other. Various τ_E scaling can be chosen, e.g. (1) INTOR-ALCATOR/Neo-ALCATOR scaling for electrons and neoclassical scaling for ions, (2) Mirrnov scaling, (3) ASDEX H mode scaling, etc. ASDEX H mode scaling with ignition margin 2.5 is assumed in this scoping study according to the guide line for this scoping study.

Troyon type of scaling is used for critical toroidal beta. Fast α and NBI particle pressures are estimated by using analytical solution of the Fokker-Plank equation (the lowest order solution of Legendre expansion) [2], [3].

Space distributions of plasma parameters can be taken into account by the following formulae.

$$f(r) = f_0 \left\{1-\left(\frac{r}{a_p}\right)^n\right\}^m$$ for density, temperature and toroidal current.

$$f(r) = f_0 + (f_a - f_0)\left(\frac{r}{a_p}\right)^n$$ for elongation.

Where, f_0 : value at plasma center

f_a : value at plasma boundary

a_p : plasma minor radius.

The parabolic distribution is assumed for plasma temperature and the distribution represented with n = 2, m = 0.3 for plasma density.

One turn plasma resistance, R_{es}, is calculated by using the following relation.

$$R_{es} I_p^2 = \int_V \eta_p(r) i_p^2(r) dv$$

Where, $\eta_p(r)$: plasma resistivity distribution

$i_p(r)$: plasma toroidal current distribution

I_p : plasma total toroidal current

Plasma resistivity, $\eta_p(r)$, is calculated by using plasma temperature profile. Classical resistivity or neo-classical resistivity can be chosen by option. In this scoping study, classical resistivity is used. The OH flux consumptions due to plasma resistance are calculated for start up and burn phase according to the assigned operation scenario. The OH flux consumption at break down phase is estimated by using the empirical scaling [4].

Main input parameters for plasma design calculation are plasma temperature, elongation, triangularity, operation scenario and various engineering parameters such as blanket/shield thickness, maximum fields and current densities in TF/PF coils, etc.

Upper and lower plasma elongations, triangularities and null points are defined in this systems code for plasma equilibrium calculations based on the assigned elongation, triangularity and the option on plasma configuration such as limiter, single null divertor and double null divertor configuraions. In single null divertor case, lower elongation, κ, and triangularity, δ, (null point side) are set to be larger than the average ones by 20 % and upper κ and δ be smaller than the everages by 20 %.

The thickness of blanket and shield can be modified from input thickness according to the neutron wall loading in calculating plasma parameters.

The required input power for plasma heating and current drive are also calculated in this plasma design step. As for the plasma heating power, the required heating power is defined as 1.5 times of the saddle point power in the equi-contour map.

1.2 Structure design

Radial and vertical builds are defined based on the results of plasma design calculation. The prohibited regions against poloidal field coil arrangement are determined from the radial and vertical builds. There are three options in determing these prohibited regions, i.e. (1) no restriction, (2) prohibited region for radial maintenance access and support of torus, and (3) prohibited region for oblique maintenance access and support of torus.

Poloidal contour of toroidal field coils is represented with one straight line (wedge part of TF coil) and three arcs in this code. The volume and weight of TF coil shear panels are calculated based on the configuration of TF coils and the prohibited regions against the PF coil arrangement.

Poloidal field coils are located at the effective positions along the line outside the TF coil contour, based on plasma equilibrium calculations. Cross-sectional sizes of PF coils are determined so that the current density in each PF coil does not exceed the maximum allowable value. When the minimum distance between some PF coil and TF coil is shorter than the assigned one, this PF coil is moved outwards so as to keep the assigned distance from TF coils.

Cross-sectional view of the first wall, blanket, shield and vacuum vessel are defined so as to match with a plasma configuration and TF coil shape.

1.3 Magnetic field calculation

The scenario for time dependent magnetic flux consumption is defined by a plasma current ramp up scenario (input data) and resistively consumed magnetic flux obtained in the step of plasma design. Poloidal field coil currents are provided by summing up two components, i.e. those (1) for plasma equilibrium and (2) for OH flux supply, according to the scenario of magnetic flux consumption. In superposing the latter component on the former one, magnetic fields in all of PF coils are calculated. The time dependent PF coil current patterns are calculated so that the peak magnetic field in all of PF coils are determined to be less than the maximum allowable field (input data). This procedure provides a maximum burn time and/or an available OH flux change. If the burn time and/or available OH flux change is significantly different from the assumed burn time/available OH flux change supposed in calculating plasma parameters, calculation must be redone all over again.

1.4 Others

Based on the time dependent PF coil current patterns, the power supply requirements are calculated such as power supply capacity, maximum poloidal magnetic store energy, peak power, etc.

The cooling system is assumed to consist of several subsystems : (1) divertor/limiter plate cooling subsystem, (2) first wall/blanket cooling subsystem, (3) shield cooling subsystem, and (4) plasma heating/current drive system cooling subsystem. The mass flow rate of each subsystem is calculated based on heat load, inlet/outlet temperatures and other data.

The characteristic parameters are calculated for the exhaust system (required pumping speeds, etc.), cryogenic system (cryogenic loads, etc.) and tritium system (tritium consumption/circulation rates, etc), based on the date such as plasma densities, fusion power, neutron wall loading and so on.

The capital costs are calculated by using the characteristic parameters of each system. The cost of land, buildings and plasma diagnostics are assumed to be constant independently of the reactor design.

2. Results of INTOR Scoping Studies

The parametric surveys are performed to study the following items (1) effect of plasma shape on burn time, (2) comparison between single null and double null divertor, (3) effect of increased peak toroidal field, (4) dependence on plasma shape (elongation and triangularity), (5) comparison between radial access and oblique access, (6) comparison between inductive and quasi steady state operations, (7) ergodic limiter, and (8) steady state current drive operation. In these studies, the peak toroidal field and overall current density in TF coils are assumed as 12 T and 13 A/mm², respectively, except the survey of (3). Plasma temperature is fixed to 10 keV except the survey of (8). Safety factor in cylindrical model, q_I, is fixed to 2.1 at plasma boundary. The capital costs are normalized by that of standard case in these studies. Main features of the standard case are as follows.

Plasma configuration	Single null divertor
	$\kappa = 1.8$, $\delta = 0.35$
Reactor maintenance	Radial access
Operation scenario	Quasi steady state
Burn time	1500 sec
Peak toroidal field	12 T
Overall current density in TF coils	13 A/mm²
Peak poloidal field	$\lesssim$ 10 T
Current density in PF coils	$\lesssim$ 25 A/mm²

The followings are the cost percentages of subsystems in the case of the stndard reactor.

Torus system	25 %
Toroidal coil system	16 %
Poloidal coil system	13 %
Heat transfer and cryogenic cooling systems	6 %
Plasma heating and current drive system	12 %
Power supply system (quasi steady state operation)	8 %
Exhaust and tritium systems	8 %

Instrumentation and control (including diagnostics)	2 %
Others (including land and buildings)	10 %

The results of these parametric studies are briefly described in the following paragraphs.

2.1 Effect of plasma elongation and triangularity on burn time

A possible burn time is compared between inductive and quasi steady state operation scenarios as functions of plasma elongation and triangularity. Figure 1 shows the difference in a possible burn time between the two operation scenarios, $\Delta\tau_{burn}$. A single null divertor configuration is assumed and peak magnetic fields in PF coils are limited to less than 10 T. The possible burn time difference, $\Delta\tau_{burn}$, decreases with the increases of plasma elongation and triangularity. The possible burn time of inductive operation case, τ_{burn}^{Ind}, is set to ~ 500 sec in this figure. The check calculations are also carried out to survey the τ_{burn}^{Ind} dependence by changing τ_{burn}^{Ind} from ~50 sec to ~700 sec and the results show that the $\Delta\tau_{burn}$ dose not strongly depend on τ_{burn}^{Ind} (error is approximately 5 ~ 10 %). The large difference of the possible burn time can be expected between inductive and quasi steady state operation in the machine with a relatively low plasma elongation and triangularity. On the other hand, only a small difference in $\Delta\tau_{burn}$ is expected in the machine with a high plasma elongation and triangularity.

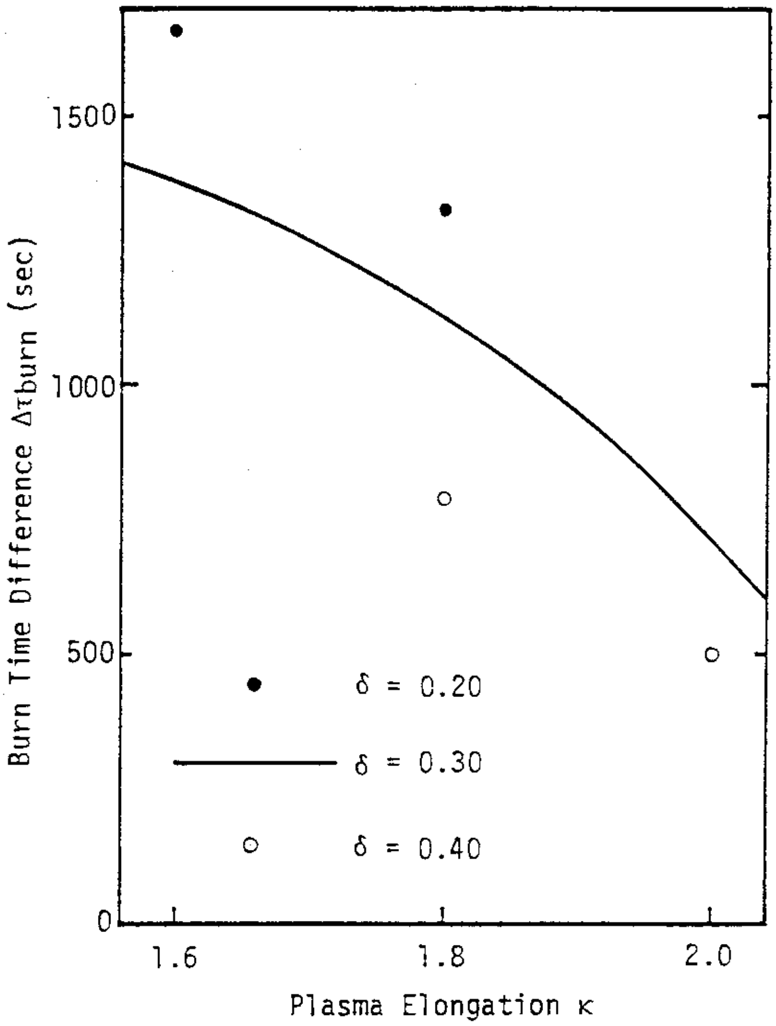

Fig. 1 Difference of possible burn time between quasi steady state operation and inductive operation as functions of plasma elongation and triangularity.

2.2 Comparison Between Single Null and Double Null Divertor Configurations.

The comparative study is carried out between single null and double null divertor cases. No restriction is imposed on PF coil locations and quasi steady state operation is adopted in these calculations. Burn time is set to approximately 1500 sec. The dependences of the key output parameters on plasma elongation, κ, are shown in Figs. 2(a) and 2(b). Figure 2(a) shows the dependences of plasma major and minor radii R_p, a_p, and total machine weight on plasma elongation. Figure 2(b) shows the dependences of the relative capital cost, magnetic stored energy and total Ampere turn of PF coils. All of these output parameters decrease monotonically with the increase of a plasma elongation, κ, within the region of $1.6 \lesssim \kappa \lesssim 2.0$. These output parameters of single null divertor case are always less than those of double null divertor case except plasma major and minor radii, though no restriction is imposed on PF coil arrangement.

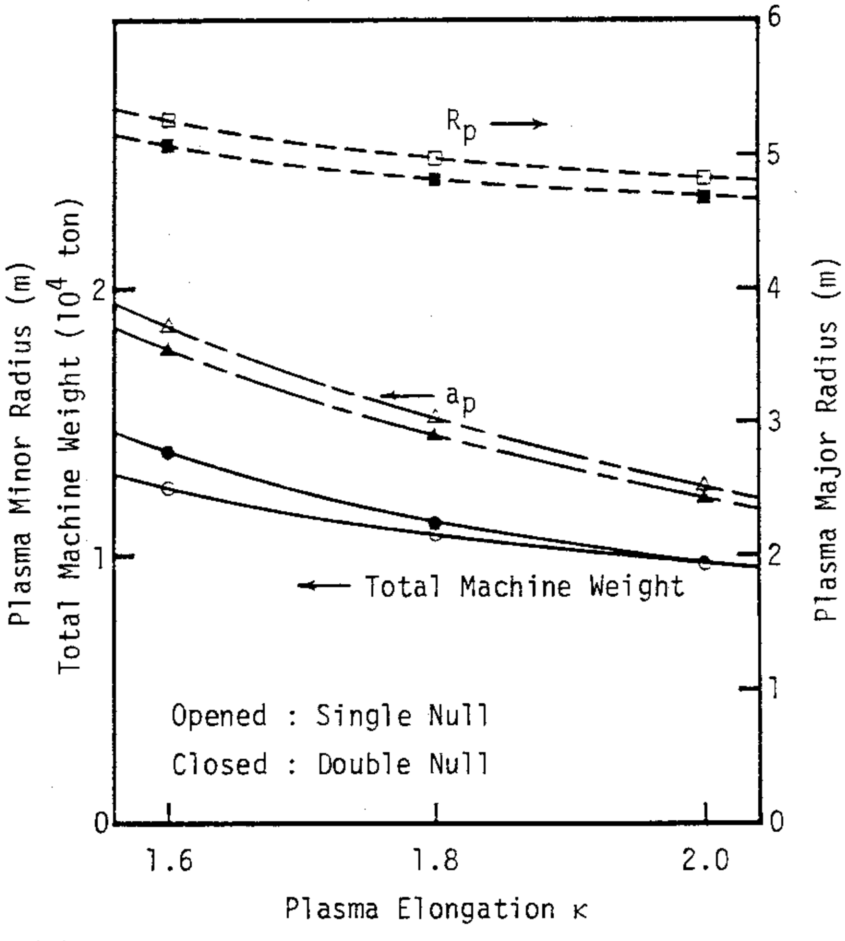

Fig. 2 (a) Dependences of plasma major radius, R_p, minor radius, a_p, and total machine weight on plasma elongation for both single null divertor and double null divertor cases.

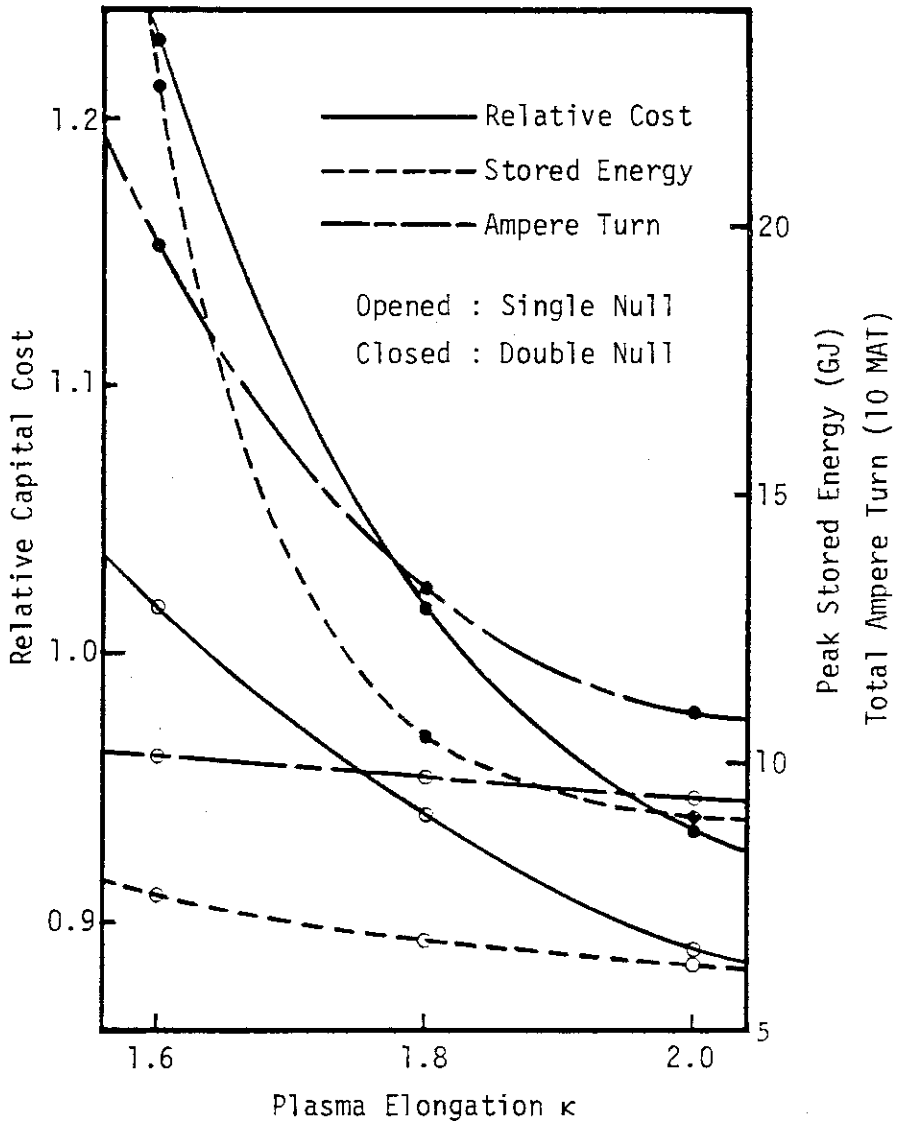

Fig. 2 (b) Dependences of the relative capital cost, magnetic stored energy and total Ampere turn in PF coils on plasma elongation for both single null divertor and double null divertor cases.

2.3 Effect of High Magnetic Field in TF Coil

The merit of increasing peak magnetic field in TF coils, B_{tp}, is surveyed for a single null divertor, quasi steady state operation case. No restriction is imposed on PF coil locations and burn time is fixed at approximately 1500 sec again in these calculations. The overall current density in TF coils, J_{cav}, is assumed to decrease so as to keep B_{tp} x J_{cav} constant in this survey.

The results are summarized in Figs. 3(a), 3(b) and 3(c). Figure 3(a) shows the effect of increasing B_{tp} on plasma major radius, R_p, minor radius, a_p, and summation of major and minor radii, $R_p + a_p$. A plasma major radius increases with the peak magnetic field, B_{tp}, though a minor radius decreases. The reduction in a minor radius increases a plasma resistance and then increases a major radius to obtain the same burn time as assigned. Figure 3(b) represents the dependences of the relative capital cost, peak poloidal magnetic stored energy and peak total ampere turn on the peak magnetic field, B_{tp}. The effect of increasing B_{tp} on these parameters is very small as shown in this figure. Figure 3(c) shows the B_{tp} dependence of total machine weight, TF coil weight and torus weight. It can be seen that the cost reduction due to the machine size reduction is compensated by the cost increase in TF coils.

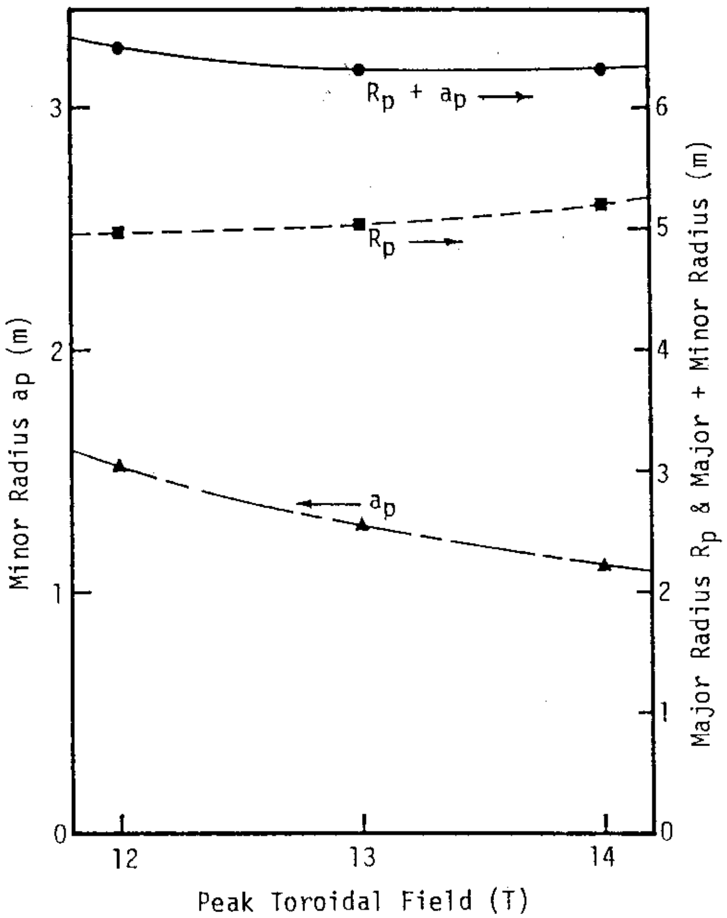

Fig. 3(a) Effects of increasing peak toroidal field, B_{tp}, on plasma major radius, R_p, minor radius, a_p, and $R_p + a_p$.

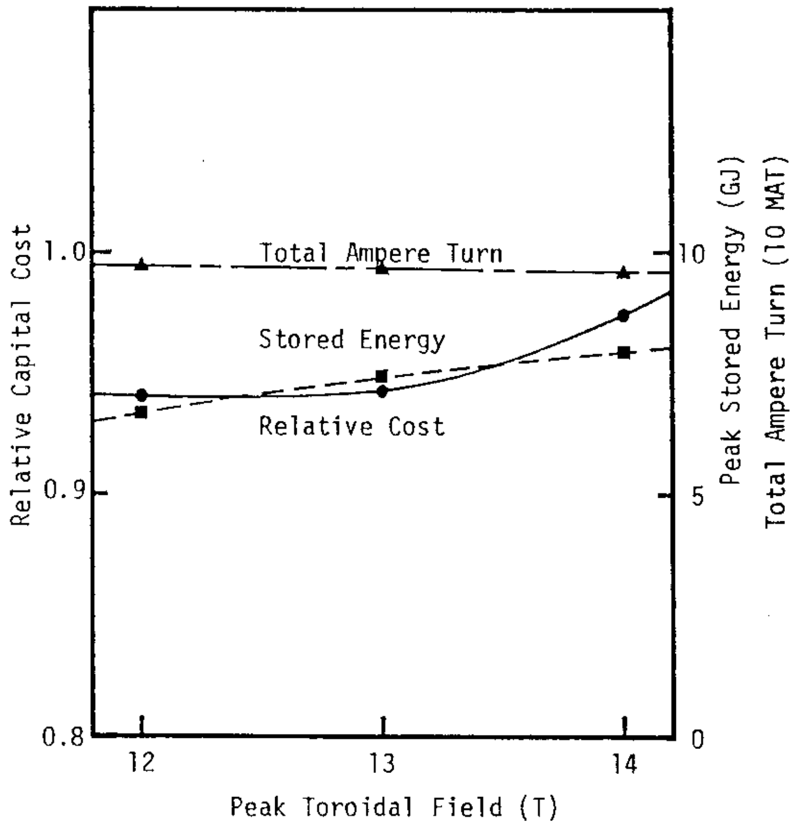

Fig. 3(b) Effects of increasing peak toroidal field, B_{tp}, on the relative capital cost, magnetic stored energy and total Ampere turn in PF coils.

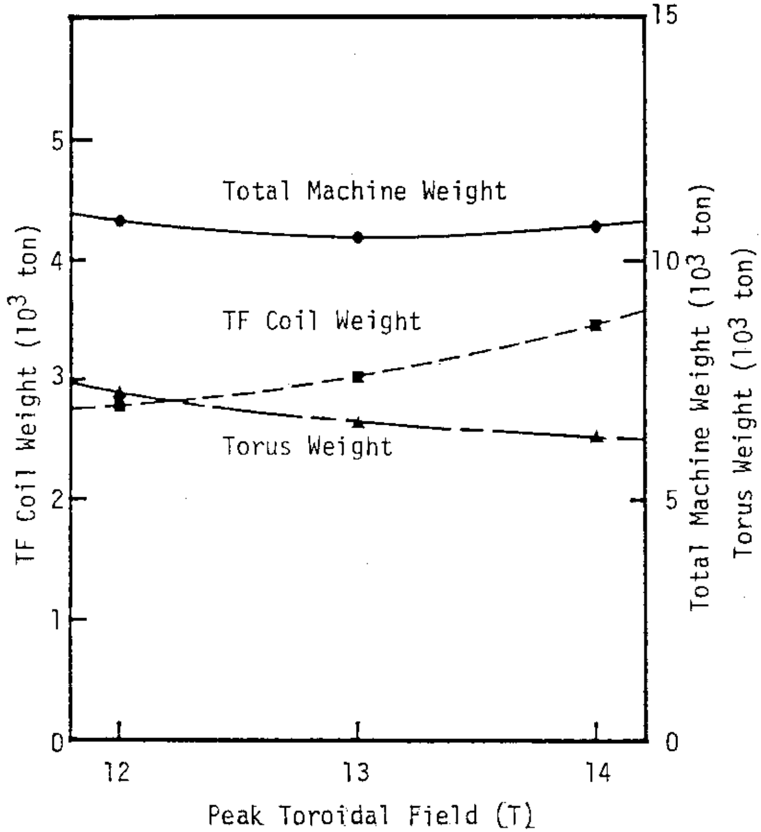

Fig. 3(c) **Effects of increasing peak toroidal field, B_{tp}, on total machine weight, TF coil weight and torus weight.**

2.4 Effect of Plasma Elongation and Triangularity

The effect of a plasma elongation, κ, and triangularity, δ, is studied for a single null divertor, quasi steady state operation case. The radial access is assumed for reactor maintenance. Then, the prohibited regions are imposed on PF coil locations considering a radial remote maintenance and support of reactor. A burn time is fixed at ~1500 sec again in this survey.

Figures 4(a) and 4(b) show the results of this survey. There is a weak minimum point in the relative capital cost as shown in Fig. 4(a). This result comes from the cost increase of PF coil system and power supply. The peak stored energy and total ampere turn of PF coils increase with elongation, κ, (compare with Fig. 2(b)). The weight of PF coils also increases with κ, though total weight decreases as shown in Fig. 4(b). The relative capital cost decreases with the increase of δ, however the sensitivity is weak.

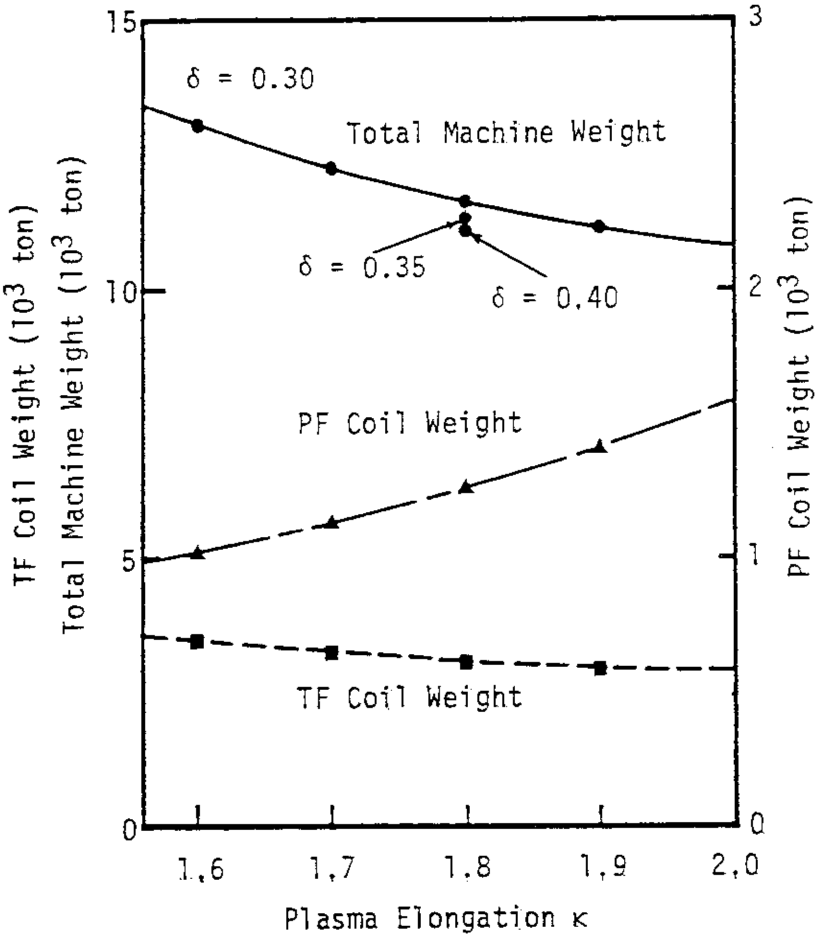

Fig. 4 (a) The relative capital cost, magnetic stored energy and total Ampere turn in PF coils as functions of plasma elongation and triangularity.

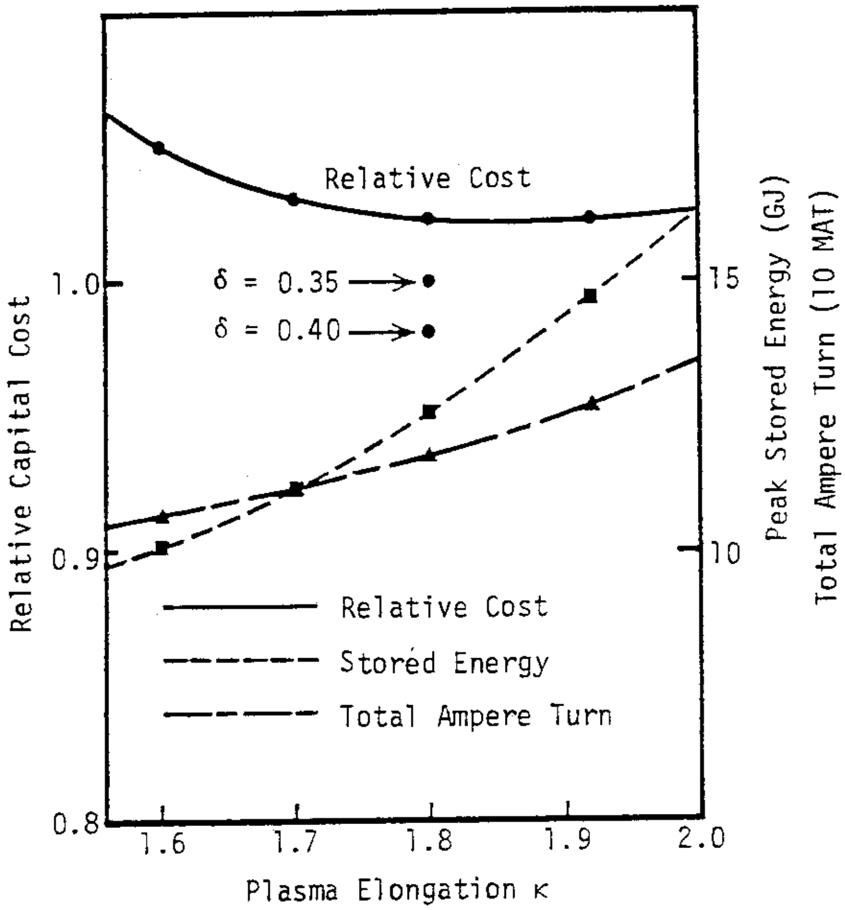

Fig. 4(b) Total machine weight, TF coil weight and PF coil weight as functions of plasma elongation and triangularity.

2.5 Comparison between Radial Access and Oblique Access

The impact of adopting oblique access for reactor maintenance is studied by comparing with the radial access case. A plasma configuration is a single null divertor with $\kappa = 1.8$ and $\delta = 0.35$. A quasi steady state operation is assumed and a burn time is fixed at ~1500 sec in both cases.

The result is summarized in Table 1. The reduction of the relative capital cost due to the oblique access is only ~3 %, though the peak stored energy of PF coils decreases by ~30 %.

Figure 5(a) and 5(b) show the cross-sectional views of radial access type and oblique access type of reactors, respectively.

Table 1 Comparison of Oblique Access
($\kappa = 1.8$, $\delta = 0.35$)

	Radial Access	Oblique Access
Plasma Major Radius (m)	4.97	5.01
Plasma Minor Radius (m)	1.52	1.50
Plasma Current (MA)	10.9	10.7
Toroidal Field (T)	4.43	4.51
Total Weight (ton)	11350	10990
TF Coil Weight (ton)	3030	2750
PF Coil Weight (ton)	1150	1020
Peak Stored Energy (GJ)	11.65	7.84
Power Capacity (MVA)	787	675
Total Ampere Turn (MAT)	115	101
Relative Capital Cost	1.0	0.970

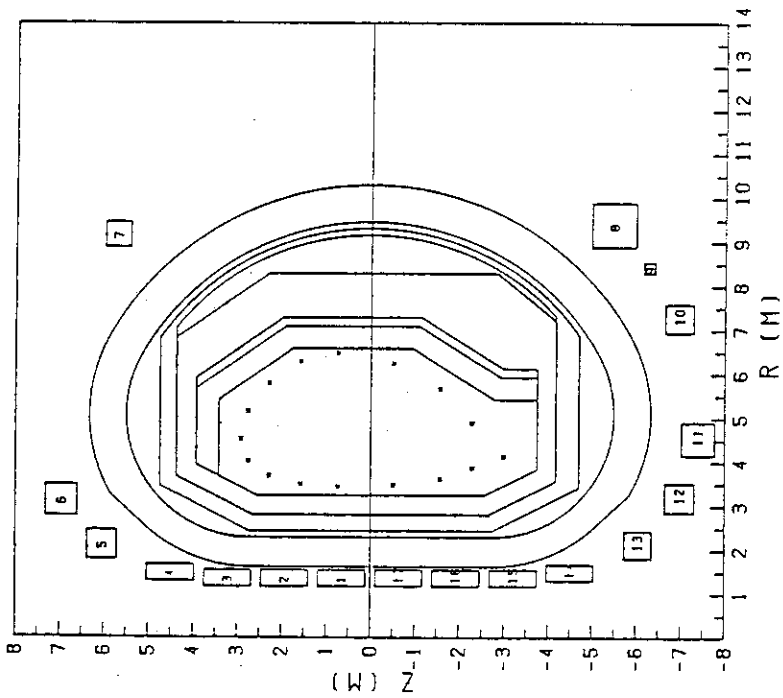

Fig. 5(a) Cross sectional view of radial access type reactor.

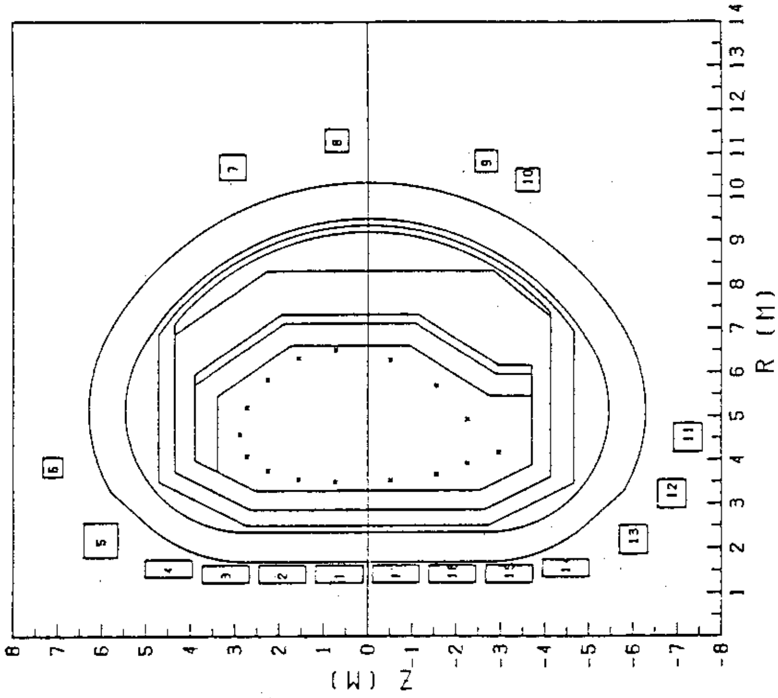

Fig. 5(b) Cross sectional view of oblique access type reactor.

2.6 Inductive Operation vs. Quasi Steady State Operation

The comparison study is carried out between inductive operation and quasi steady state operation scenarios as a function of plasma major radius (or burn time). The plasma configuration assumed in these calculations is a single null divertor one with elongation $\kappa = 1.8$ and triangularity $\delta = 0.35$. For the radial reactor maintenance access and torus support, the prohibited regions are defined against the PF coil arrangement. The result is shown in Figs. 6(a) and 6(b). Figure 6(a) represents the dependences of the relative capital cost and plasma minor radius, a_p, on plasma major radius, R_p. The burn time of each reactor design (each calculation point) is written near each calculation point. The plasma minor radius decreases with the increase of R_p, since the τ_E scaling used in the scoping study has a R_p dependence instead of a_p dependence and the toroidal field at plasma center increases with R_p. There are minimum points in the relative capital costs of both inductive and quasi steady state operation cases. It can be seen from Fig. 6(b) that the cost increments in the smaller R_p region come from the cost increments of PF coils and power supply system. Compared with inductive operation case, about 7 % of cost reduction comes from the cost reduction of PF coils and power supply system, however about 4 % of cost increment comes from the cost increment of plasma heating and current drive system. As a result, the relative capital cost of quasi steady state operation case is less than that of inductive operation case by approximately 2 ~ 4 %. In these calculations, the capital costs of both plasma current drive and heating system are independently counted in. If we can also use plasma current drive system for plasma heating, the relative capital cost will decrease by approximately 4 %. Figure 6(b) also shows the relative capital cost of quasi steady state operation case where both plasma heating and current drive can be performed by common heating system.

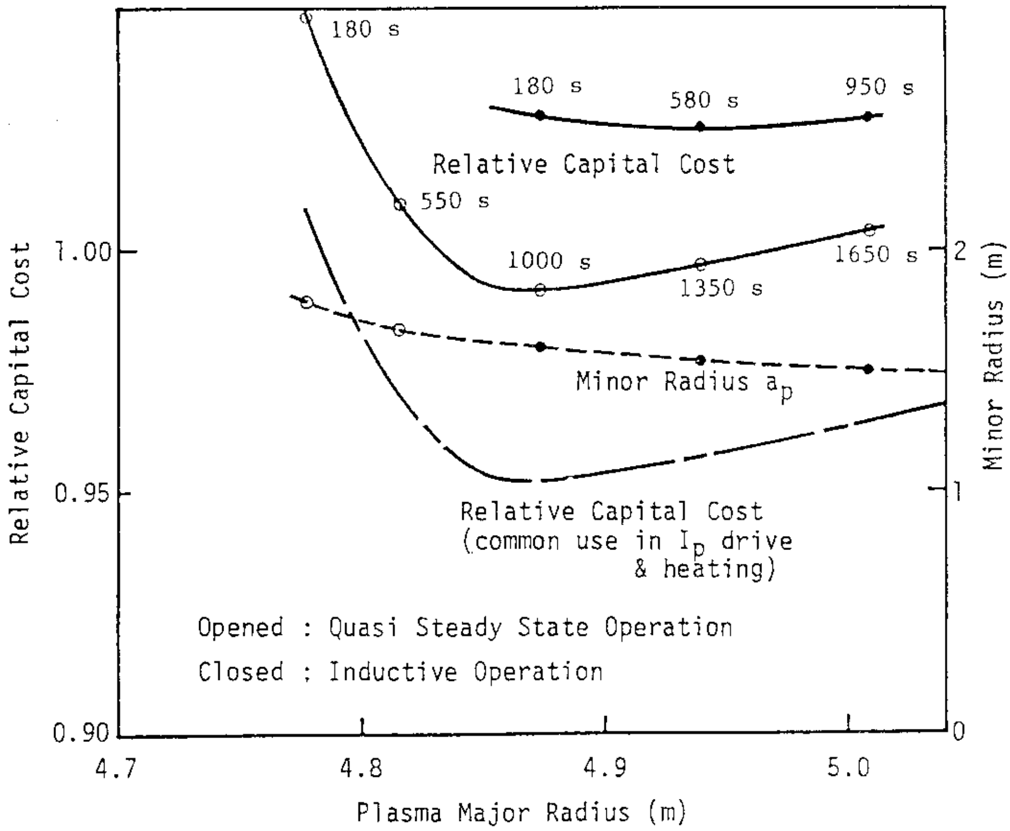

Fig. 6(a) Major radius dependences of the relative capital cost and minor radius for both inductive operation and quasi steady state operation scenario.

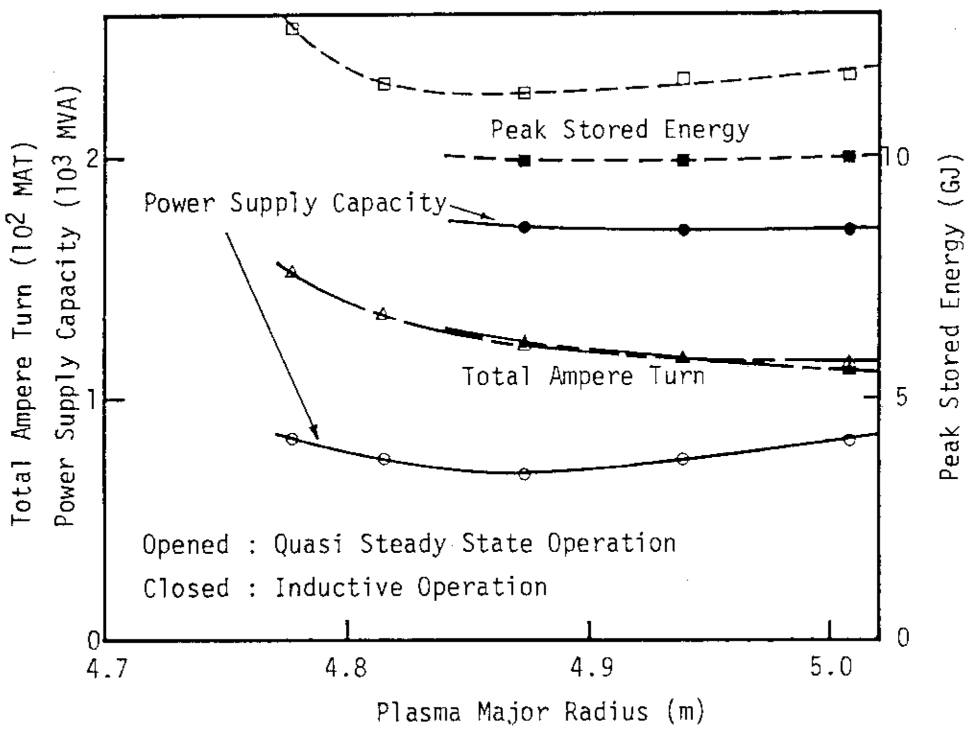

Fig. 6(b) Major radius dependences of power supply capacity (Max. Volt x Max. Ampere), magnetic stored energy and total Ampere turn in PF coils.

2.7 Ergodic Limiter

The relative merits are surveyed for systems with ergodic limiters. The thickness of the ergodic region is assumed to be 20 cm (at outboard side) ~40 cm (at inboard side). A quasi steady state operation is assumed and a burn time is set to ~1500 sec in this survey. No restriction is imposed on PF coil arrangement for simplicity. Two cases of plasma elongation are calculated, i.e. (1) κ = 1.8 and (2) κ = 2.0.

The result is summarized in Table 2. Although no restriction is imposed on PF coil arrangement, no merit could be expected by using ergodic limiter concept since the increment plasma minor radius due to ergodic plasma region increases machine size and reduces the toroidal field in the plasma center as shown in Table 2.

Table 2 Comparison of Ergodic Limiter

	Standard	Ergodic Limiter	
	κ = 1.8	κ = 2.0	κ = 1.8
Plasma Major Radius (m)	4.97	5.43	5.71
Plasma Minor Radius (m)	1.52	1.52	1.80
Plasma Current (MA)	10.9	10.8	11.7
Toroidal Field (T)	4.43	4.16	3.96
Total Weight (ton)	11350	11680	12850
TF Coil Weight (ton)	3030	3040	3340
PF Coil Weight (ton)	1150	1070	1080
Power Capacity (MVA)	787	742	692
Peak Stored Energy (GJ)	11.65	7.54	6.67
Total Ampere Turn (MAT)	115	103	97
Relative Capital Cost	1.0	1.026	1.077

2.8 Steady State Operations

The relative merits of steady state operation scenario are studied for a single null divertor configuration (RF current drive) and limiter configuration (NBI current drive). The plasma temperature is set to 20 keV in the former case, and to 10 keV and 20 keV in the latter case. The plasma elongation is fixed to 2.0 in all cases. No restriction is imposed on the PF coil arrangement for simplicity. Two cases of solenoid coil option are calculated, i.e. (1) no solenoid coil and (2) with solenoid coils. The energy multiplication factor, Q, is set to five.

The results are summarized in Table 3. The relative capital costs of steady state operation systems can be significantly reduced (10～20 %), compared with the standard case. The systems with solenoid coils are more compact, and the relative capital cost and the peak poloidal stored energy, etc. are less, compared with no solenoid coil cases. Though the 10 keV plasma is more preferable than the 20 keV plasma from the viewpoint of machine economy, a large amount of power (~200MW) is required for plasma current drive in case of 10 keV plasma. The relative merits of steady state operation scenarios are significant from machine economic point of view. However, low energy multiplication factor, Q, and low neutron wall loading are not necessarily attractive as a mission of the next generation tokamaks.

Table 3 Comparison of Steady State Current Drive

(κ = 2.0)

	Standard	NBI Steady State Current Drive with Limiter				RF steady State Current Drive with Divertor	
	κ = 1.8	No Solenoid Coil		With Solenoid Coils		No Solenoid Coil	With Solenoid Coils
		T=10keV	T=20keV	T=10keV	T=20keV	T=20keV	T=20keV
Plasma Major Radius (m)	4.97	4.04	4.55	4.17	4.53	4.13	4.25
Plasma Minor Radius (m)	1.52	1.47	2.00	1.25	1.63	1.55	1.32
Plasma Current (MA)	10.9	11.3	17.0	9.32	13.7	12.0	9.98
Toroidal Field (T)	4.43	3.45	3.06	4.16	3.83	3.37	4.08
Total Weight (ton)	11350	7790	10090	7110	8980	8980	7890
TF Coil Weight (ton)	3030	2200	2850	2110	2560	2340	2240
PF Coil Weight (ton)	1150	740	510	410	500	1070	480
Heating Power (MW)	79	184	90	185	93	78	80
Wall Loading (MW/m)	1.124	0.329	0.229	0.384	0.289	0.409	0.477
Peak Stored Energy (GJ)	11.65	7.24	6.61	2.39	3.81	8.48	3.66
Power Capacity (MVA)	787	282	280	47	76	33	104
Total Ampere Turn (MAT)	115	107	76	83	85	104	83
Relative Capital Cost	1.0	0.878	0.883	0.812	0.821	0.888	0.779

3. Japanese Proposal

The result described in the paragraph 2.2 shows that the capital cost of a single null divertor case is less than that of a double null divertor case as well as the stored energy and total ampere turn in PF coils in the range of $\sim 1.6 \lesssim \kappa \lesssim \sim 2.0$. So, we propose to adopt a single null divertor configuration rather than a double null divertor one.

Since the impact of changing a peak toroidal field, B_{tp}, is very small as shown in the paragraph 2.3, we dare not alter it from the current value 12 T. On the other hand, since the merit is clear to increase an overall current density in TF coils J_{cav}, we increase the J_{cav} to 13 A/mm² at B_{tp} = 12T based on the preliminary engineering consideration for the innovative TF coil design.

A plasma elongation should be determined not only by cost analyses but also by physical considerations on plasma vertical position control. The analyses by rigid and deformable plasma models show that a high plasma elongation beyond $\kappa = 1.8$ would be very serious in the next generation tokamak reactors

Furthermore, the relative capital cost has a weak minimum point at the plasma elongation of 1.8～1.9 in the case of a radial access for reactor maintenance as shown in the paragraph 2.4. Based on these results, we propose to select a plasma elongation of ～1.8.

The comparison study between radial and oblique access for reactor maintenance shows that the difference in the relative capital cost is only ～3 % between the two cases as shown in the paragraph 2.5. This amount of difference can not be strong motivation to adopt the oblique access at the risk of the difficulty in a reactor maintenance brought by the oblique access.

The system analysis represented in the paragraph 2.6 indicates that the relative capital cost in the case of a quasi steady state operation is less than that in the case of an inductive operation by approximately 3～7 %. Furthermore, a quasi steady state operation scenario should be adopted as the first step to a steady state tokamak reactor plant since the pulse operation of tokamak reactors is one of the big weak points. The parametric study on the effect of plasma major radius, R_p, (or burn time, τ_{burn}) shows that the relative capital cost is minimized at R_p = 4.85～4.90 m (or $\tau_{burn} \approx 1000$ sec) as described in the paragraph 2.6. Based on these considerations, we propose to adopt a quasi steady state operation scenario with ~1000 sec burn time.

The merit of an ergodic limiter would be poor as shown in the paragraph 2.7, since the ergodic region substantially increases machine size and decreases a toroidal magnetic field at plasma center. So we abandon to use an ergodic limiter concept.

The substantial cost down (10～20 %) can be expected by using steady state operation scenario (see the paragraph 2.8). However, low energy multiplication factor, Q, (Q=5) and low neutron wall loading, P_{wn}, ($P_{wn} \approx 0.2 \sim 0.4$ MW/m²) are not so attractive to adopt this concept as the next generation tokamak reactor.

Our typical proposal for the INTOR innovative reactor is summarized in Table 4. Figures 7(a), 7(b) and 7(c) show the cross-sectional view of our reactor and plasma configurations at high and low beta phases, respectively. The shaded zones in Fig. 7(a) are the PF coil prohibited regions for reactor maintenance and machine support. Figure 7(d) shows radial build of this reactor.

Table 4 Characteristics of Japanese Proposal Reactor

Plasma major radius (m)	4.87
Plasma minor radius (m)	1.60
Aspect ratio	3.05
Elongation	1.8
Triangularity	0.35
Plasma temperature (kev)	10
Electron density ($1/m^3$)	1.27×10^{20}
Ion density ($1/m^3$)	1.17×10^{20}
Fuel density ($1/m^3$)	1.11×10^{20}
Z_{eff}	1.5
Plasma current (MA)	11.6
Energy confinement time (sec)	2.33
Safety factor q_I	2.1
Total toroidal beta	6.13
Fuel toroidal beta	5.16
Toroidal field in plasma center (T)	4.16
Peak toroidal field (T)	12.0
Overall TF coil current density (A/mm^2)	13.0
Peak poloidal field (T)	10.0
PF coil current density (A/mm^2)	25.0
Fusion output power (MW)	655
Neutron wall loading (MW/m^2)	$\sim$ 1.08
Burn time (sec)	$\sim$ 1000
Operation scenario	Quasi steady state
Reactor maintenance access	radial access

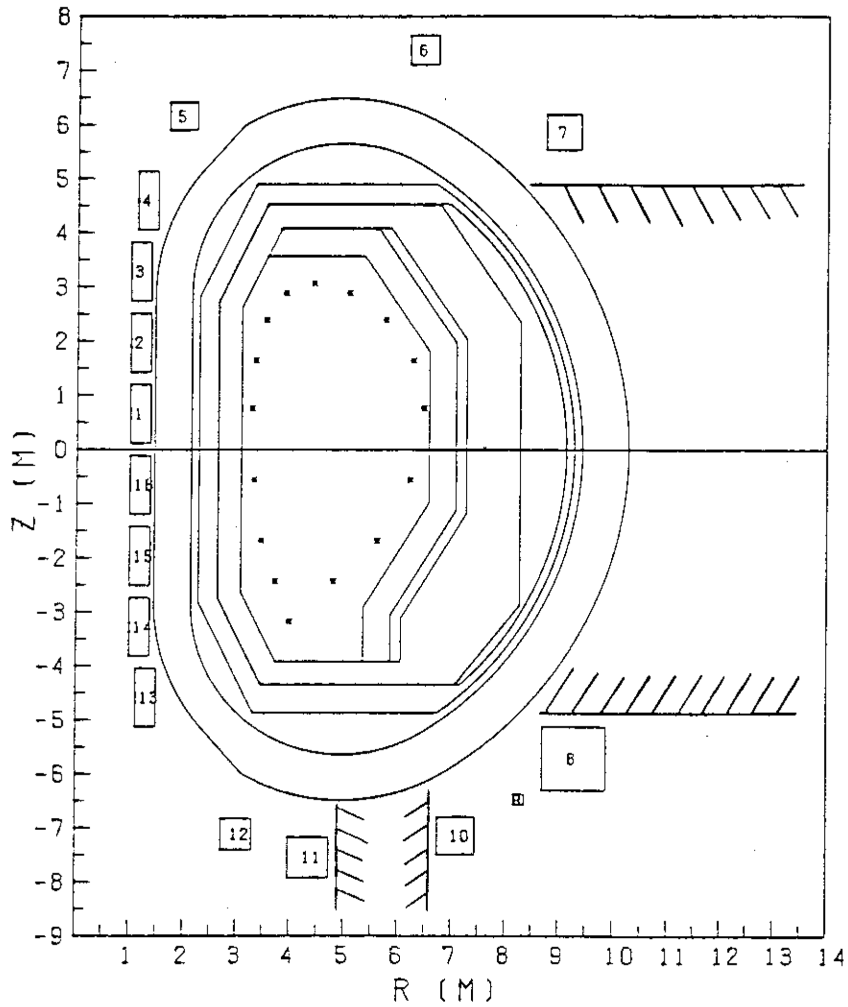

Fig. 7(a) Cross sectional view of proposed reactor.

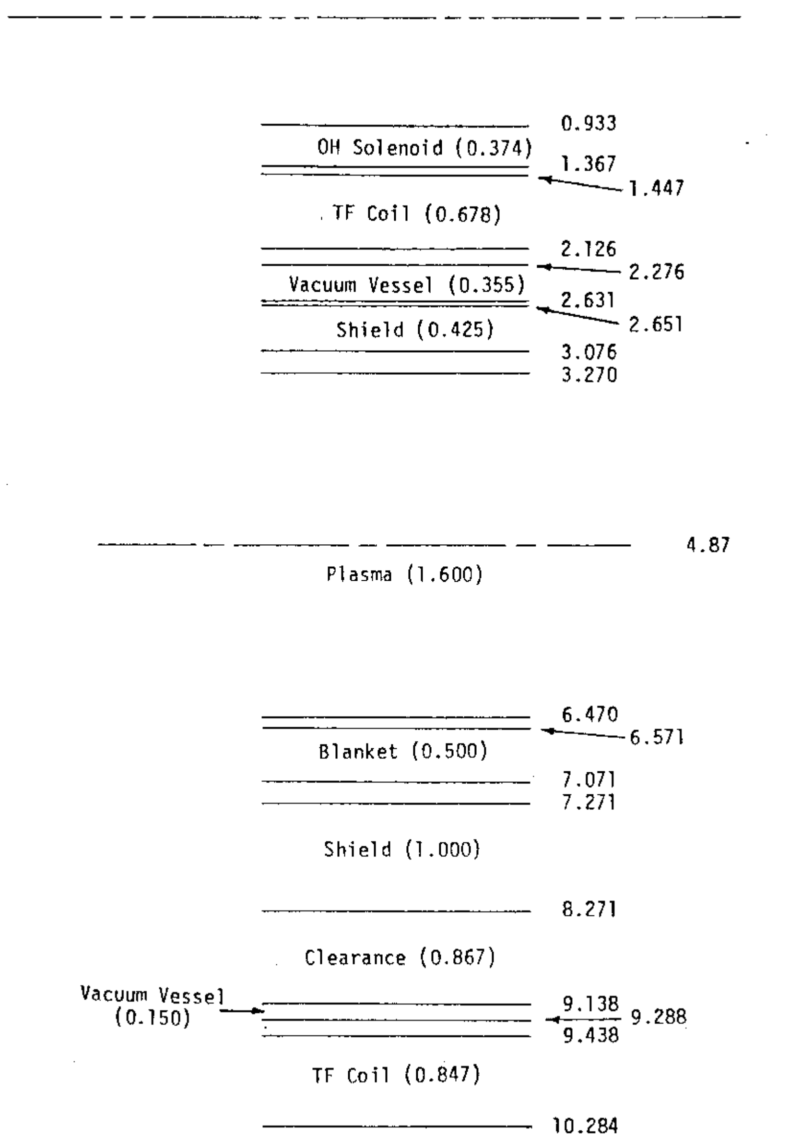

Fig. 7(d) Radial build of proposed reactor.

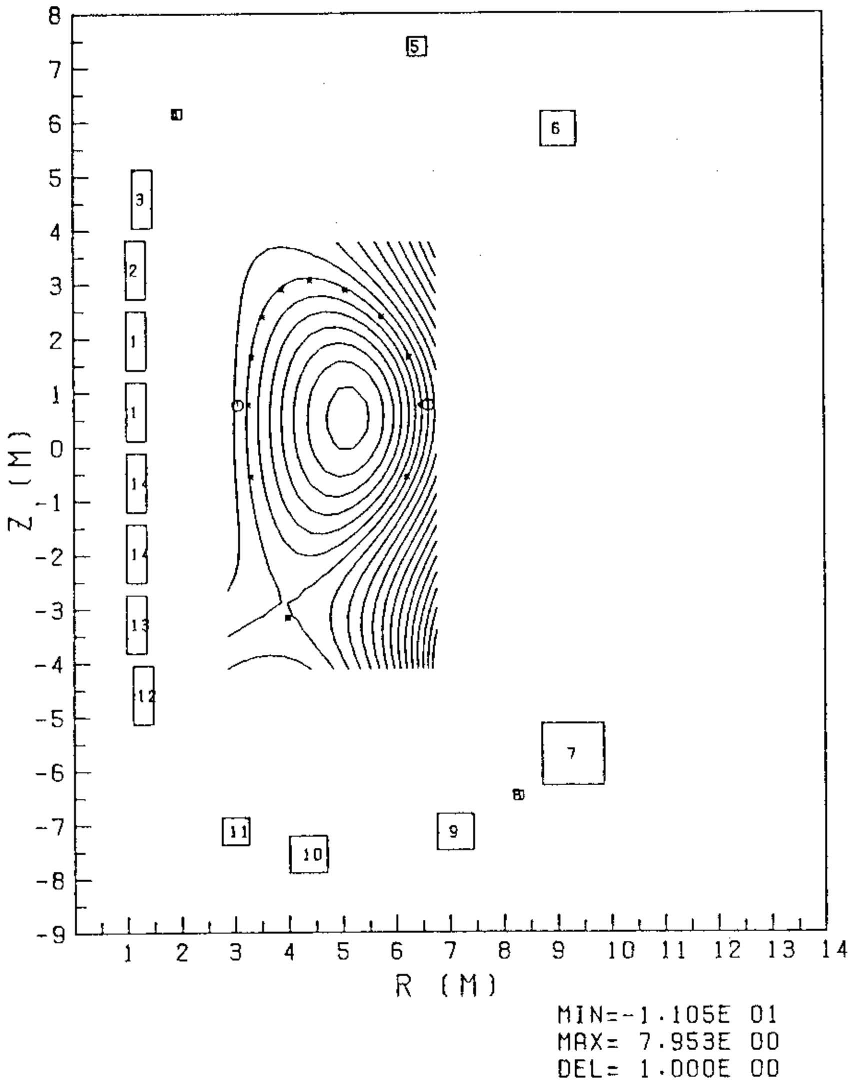

Fig. 7(b) Plasma equilibrium configuration of proposed reactor at high beta phase.

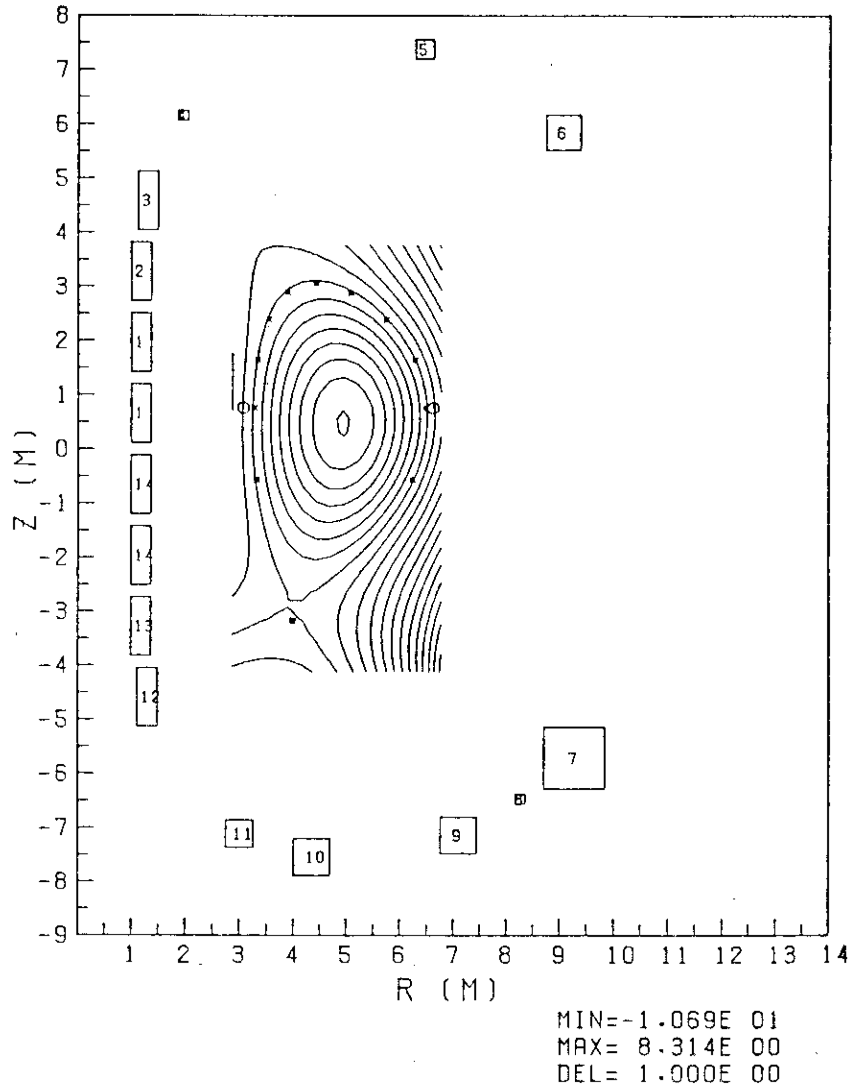

Fig. 7(c) Plasma equilibrium configuration of proposed reactor at low beta phase.

4. Summary

The results of the INTOR scoping study are summarized as follows.

(1) The difference of the possible burn time between quasi steady state and inductive operation decreases with the increases of plasma elongation and triangularity.

(2) The comparison study between single null and double null divertor configurations shows that the peak poloidal stored energy, total PF coil Ampere turn and the relative capital cost are less in single null divertor case than in double null divertor case in the range of plasma elongation $\kappa \lesssim 2.0$.

(3) The impact of increasing peak toroidal field can be expected to be very small.

(4) The capital cost has a weak minimum point at the plasma elongation of 1.8~1.9 in case of radial reactor maintenance access. In this survey, the PF coil locations are limited so as not to violate the zones for reactor maintenance and machine support.

(5) The difference of the relative capital cost is estimated to be approximately 2～4 % between inductive operation and quasi steady state operation scenarios (Inductive operation case is expensive.).

The capital costs of both inductive operation and quasi steady operation cases have a minimum point at some plasma major radius (or some burn time). The increase of cost in the smaller major radius region is due to the cost increments of PF coils and power supply system.

(7) No clear merit can be expected by using ergodic limiter concept since the ergodic region increases the machine size and reduces the toroidal field at the plasma center.

(8) The capital cost can be significantly reduced (10～20 %) by adopting steady state operation scenario. However, the low energy multiplication factor and low neutron wall loading would be problem of this scenario.

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