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Systems analysis code named NEW-TORSAC (TOkamak Reactor Systems Analysis Code) has been developed by modifying the TORSAC which had been already developed by us. The NEW-TORSAC is available for tokamak reactor designs and evaluations from experimental machines to commercial reactor plants. It has functions to design tokamaks automatically from plasma parameter setting to determining configurations of reactor equipments and calculate main characteristics parameters of auxiliary systems and the capical costs. In the case of analyzing tokamak reactor plants, the code can calculate busbar energy costs.

Some output of this code such as a reactor configuration, plasma equilibrium, electro-magnetic forces, etc. are graphically displayed as well as numerical output.

The code has been successfully applied to the scoping studies of the next generation machines and commercial reactor plants.

Keywords : Systems Analysis, NEW-TORSAC, Plasma Parameter Setting, Automatic Design, Experimental Reactor, Power Reactor Plant, Capital Cost, Busbar Energy Cost

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トカマク炉システムの解析・評価コード"NEW−TORSAC"の開発

日本原子力研究所那珂研究所臨界プラズマ研究部
笠井 雅夫*・井田 俊雄*・西川 正名*
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（1987年6月22日受理）

原研と三菱原子力工業（株）との協同研究として，トカマクシステムコード(NEW−TORSAC)が開発された。本コードは実験炉から実用炉まで適用が可能で，プラズマパラメータの設定からトカマク装置の設計までを自動的に行なうことができ，補器類の特徴的なパラメータを計算し，炉の建設費を概算することが可能である。また動力炉の解析では発電コストも求めることができる。本コードはINTORのスコーピングスタディおよび動力炉のパラメトリックスタディに適用され，その有効性が確認された。

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1. Introduction

Importance of the system studies for the fusion reactors have been increasing. Scale of the next generation fusion systems has rapidly increased and would exceed that of fission power plant in some respects such as weight and cost. Therefore, system optimization should be carried out for the rational research and development. On the other hand, accumulation of the knowledges on the fusion reactors and rapid development of computers have made it possible to carry out an effectual system optimization.

We have started to develop a system analysis code at the end of 1979 and the final version of the code : TORSAC (TOkamak Reactor System Analysis Code) was completed at the summer of 1981 [1]. Although several stages of improvements [2] have been applied, the TORSAC is characterized by two basic functions : (1) generation of new tokamak design based on the reference design, and (2) computation of design characteristics of the generated design. As easily understandable from above two functions, the TORSAC is essentially a tool for rapid design and evaluation of tokamak systems. User of the TORSAC has to specify, at first, an appropriate reference design and a way of design modification to generate a target system design. Second task of the user is to optimize the design according to the resultant design characteristics. User oriented system optimization described above was chosen from the reason that the code was prepared for any kind of tokamak from a small experimental device to a commercial reactor. From this reason, it was difficult to implement a specific procedure of system design and optimization.

During these several years, plasma physics has been developed and an interest has been focussed on D-T fusion reactors. According to these changes in situations, fundamental improvement of the TORSAC to the NEW-TORSAC was planed at march 1986. Here, an object is restricted to the D-T fusion tokamak systems, from experimental to commercial reactors, using super-conducting magnets. Further, automatic design and optimization algorithm are implemented in the program. In other words, a lot of design knowledge which have been acquired by designer has been transfered to the NEW-TORSAC and remained task for system designer is to apecify basic parameters as input data.

The NEW-TORSAC was completed at the summer of 1986 and some computational results were presented at the last INTOR workshop [3]. Small scale improvements has been continued but the main frame of the NEW-TORSAC has been fixed. Therefore, structure and capability of the code is presented here. The results of the INTOR scoping study [3] by the NEW-TORSAC will be helpful to understand the capability of it.

In the section 2, main structure and characteristics of the NEW-TORSAC are described. The typical results computed by this code are shown in the section 3, and the summary of this paper is described in the section 4.

2. General Description on System Code

The TORSAC systems analysis code was revised to design and evaluate tokamak reactors (from the next generation experimental reactors to commercial reactor plants). The NEW-TORSAC consists basically of four parts : (1) plasma parameter setting, (2) tokamak system design, (3) magnetic field calculation, and (4) others. First of all, plasma parameters are calculated so that the performance objectives (e.q. ignition margin, burn time, wall loading, fusion output power, etc.) can be obtained. Then, the radial and vertical buidls of tokamak machine are defined based on the calculated plasma parameters and other necessary input data (thickness of clearances, blanket, shield, vacuum vessel, etc.). The configurations of the first wall, blanket, shield, vacuum vessel and toroidal field (TF) coils are automatically generated in this code. The optimal poloidal field (PF) coil locations are determined based on plasma equilibrium calculations with taking into account the prohibited regions against the PF coil arrangement. The operation patterns of the PF coil currents are obtained by superposing Ohmic heating (OH) flux component on plasma equilibrium component, according to the assigned and calculated (OH flux comsumption, etc.) operation scenario data. The PF coil sizes are determined so that the current densities in the PF coils do not exceed the assigned value. In determing the PF coil current patterns, the peak magnetic fields in the PF coils are calculated and are limited to less than the assigned value. This limitation on peak magnetic fields in the PF coils determines burn time and/or available OH flux change. Based on the operation patterns of the PF coil currents, the key parameters on the PF coil power supply are calculated (Max. Ampere x Max. Volt, stored energy, peak power, etc.). Other parameters such as pumping requirements, cooling requirements, requirements for tritium processing system, and cost of reactor equipments are also calculated. In Fig. 1, the flow diagram of the NEW-TORSAC is shown. In the following paragraphs, brief descriptions are provided for the main routines of this systms code.

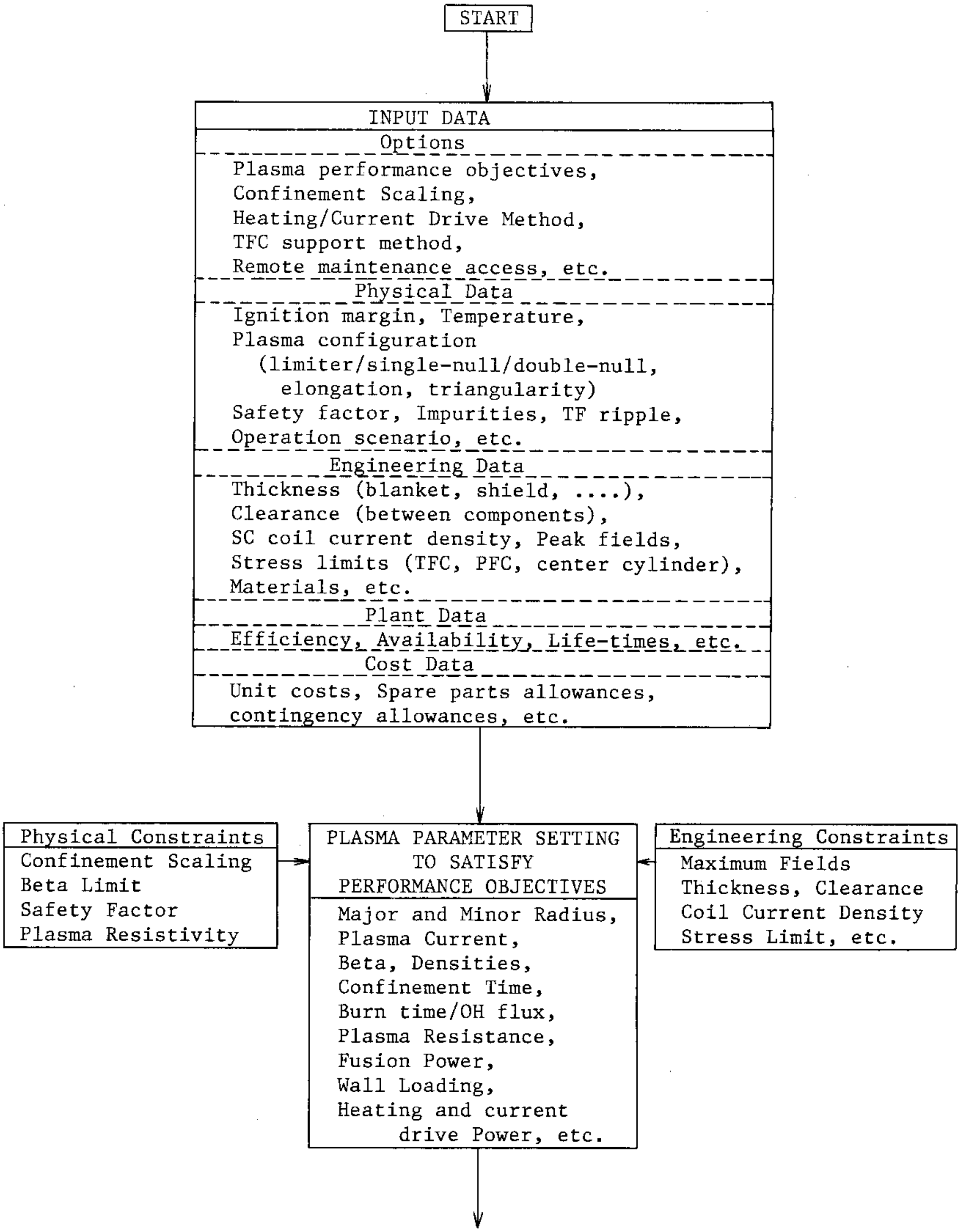


Fig. 1 Schematic Flow Diagram of the NEW-TORSAC

MACHINE CONFIGURATIONS

- Radial Build
- Vertical Build
- TF Coil Configuration (Vertical/horizontal cross-section)
- Configurations of First Wall, Blanket, Shield, Vacuum Vessel, etc.
- PF Coil Location
- PF Coil Current and Size
- Burn time
- Volumes and Weights
- Passive Property or Torus System
- Feasibility Check of Plasma Vertical Position Control, etc.

MAGNETIC FIELD CALCULATIONS (1) → (PF Coil Location / PF Coil Current and Size / Burn time)

- Plasma Equilibrium
- OH Flux Supply
- Peak Poloidal Field

↓

AUXILIARY SYSTEMS

- SC Coil Power Supply (peak power, power supply capacity, stored energy, coil voltage, etc.)
- Exhaust system (He, D, T pumping speed, etc.)
- Heating System
- Cooling System (main heat transfer, other heat transfer)
- Cryogenic System (LHe/LN_2 refrigerator load)
- Fuel Handling and Storage Systems (fuel circulating rate, tritium production rate)

↓

MAGNETIC FIELD CALCULATION (2)

- Toroidal Field Ripple
- Peak Toroidal Field
- Centering Force of TF Coils
- Hoop Force of TF Coils
- Hoop Forces of PF Coils
- Mutual Forces between PF Coils

↓

Fig. 1 Schematic Flow Diagram of the NEW-TORSAC (continued)

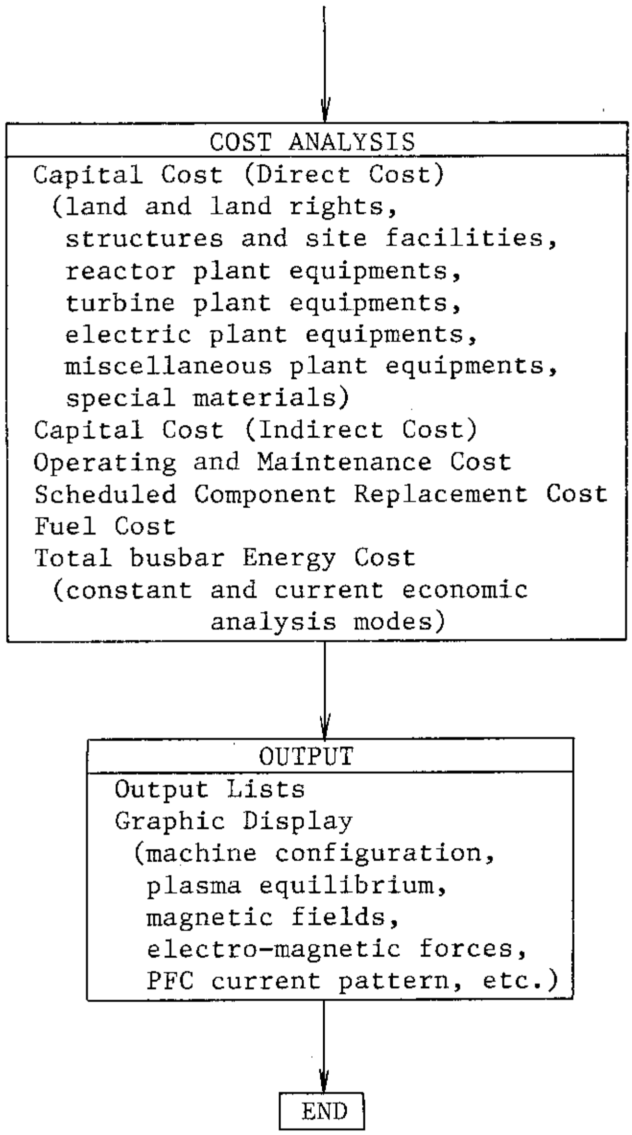


Fig. 1 Schematic Flow Diagram of the NEW-TORSAC (continued)

2.1 Plasma parameter setting

The flow diagram of the procedure to determine plasma parameters are shown in Fig. 2. Main input parameters for plasma design calculation are temperature, elongation, triangularity, operation scenario and various engineering parameters such as blanket and shield thickness, peak fields are current densities in TF and PF coils, stress limits in TF coils, TF coil casings, PF coils, and center cylinder, and so on.

The plasma temperature can be either input data or calculated in the code so as to maximize fusion power density under given beta value and toroidal field by option.

The plasma elongation κ, and triangularity, γ, are key parameters for plasma vertical instability. These parameters also affect the burn time [3]. However, it requires detailed analyses to determine the plasma shape parameters, taking into account these problems. So these shape parameters are given as input data in this code based on the design experiences.

In single-null divertor cases, upper and lower plasma elongations and triangularities are determined in the code. The elongation and triangularity of null point side (lower side in our code) are set to be larger than the averages by 20 % so that the average elongation and triangularity are equal to the input ones [4].

After plasma size (R_p, a_p) is determined, the positions of upper and/or lower null points are calculted for plasma equilibrium calculations based on the assigned elongation and triangularity.

The thickness of blanket and shield can be modified from input thickness according to the neutron wall loading in calculating plasma parameters, by using the reference neutron wall loading and the attenuation factors in the blanket and shield regions.

We prepare four representations of the safety factor [5], [6], [7] which are shown as follows.

$$q^* = \frac{\pi a_p{}^2 B_t}{\mu_o I_p R_p} \{1 + \kappa^2(1 + 2\gamma^2)\}$$

$$q_{eff} = \frac{\pi a_p{}^2 B_t}{\mu_o I_p R_p} (1 + \kappa^2)\{1 + 1/A_p{}^2(1 + \tfrac{1}{2}\Delta^2)\}f$$

$$A_p = R_p/a_p, \ f = 1.25 - 0.54\kappa + 0.3(\kappa^2 + \gamma^2) + 0.13\gamma$$

$$\Delta = \beta_p + \tfrac{1}{2}\ell_i$$

$$\frac{1}{q_J} = \frac{\mu_o I_p R_p}{2\pi a_p{}^2 B_t}(1 - 1/A_p{}^2)^{1/2}\{1/(1 + \kappa^2) - 0.08\gamma\} - 0.07\{1 + (\kappa - 1)\gamma\}$$

$$q_F = \frac{\pi a_p{}^2 B_t}{\mu_o I_p R_p}(1 + \kappa^2)(1 + f_1/A_p{}^{1/2})(1 + \beta_p f_2/A_p{}^{1.5})$$

$$f_1 = 0.16 + 0.633\gamma, \quad f_2 = 0.45 + \gamma$$

One of these safety factors can be selected by option and be fixed to given value (input data) at plasma parameter setting.

Troyon type scaling is used for determing critical toroidal beta. Energetic α and NBI particle pressures are estimated by using analytical solution of the Fokker-Plank equation (the lowest order solution of Legendre expansion) [8], [9] : the fast ion pressure P_f is estimated by the following equation.

$$P_f = S_f \tau_s E_f G_e / 3$$

Here, S_f is the energetic ion source rate, E_f is the initial energetic ion energy, τ_s is the Spitzer ion-electron momentum exchange time, and G_e is the fraction of the energy which goes to electrons.

Since the neutral beam deposition profile and shine through rate can not be calculated unless the plasma geometrical parameters (R_p, a_p) are defined, the pressure of the fast neutral beam ions is roughly estimated by the approximate method to set plasma parameters in the case of steady state operation with non-inductive current drive by neutral beam injection. In order to confirm the energetic ion pressure of the neutral beam, the neutral beam depositionprofile is numerically calculated after plasma parameter setting. If this NBI energetic ion pressure by detailed calculation is different from that obtained by rough estimation, the calculation of plasma parameter setting should be carried out again.

Plasma parameters can be chosen to satisfy one of the following combinations of performance objectives in our systems code, i.e. (1) ignition margin and burn time/available Ohmic heating (OH) flux aupply, (2) ignition margin and location of TF coil inner leg, (3) ignition margin and neutron wall loading, (4) ignition margin and fusion power, (5) burn time/available OH flux supply and fusion power, and (6) neutron wall loading and fusion power. The simplified relation is used to provide the assigned burn time or available OH flux supply in the step of plasma design. The second option of the performance objective set is prepared for analyzing reactors operated in steady state.

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.

The ignition margin, C_{IG}, is defined as $C_{IG} = Q_\alpha/Q_{loss}$, where Q_α is the α-heating rate and Q_{loss} is the total energy loss rate. The transport energy loss, the Bremstrahlung loss and the synclotron radiation loss are taken into account as the energy losses in our code. 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.

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

$$f(r) = f_o\{1 - (r/a_p)^m\}^n$$ for density, temperature and toroidal current.

$$= f_o + (f_a - f_o)(r/a_p)^2$$ elongation.

where, f_o : value at plasma center
f_a : value at plasma boundary
a_p : plasma minor radius.

As for plasma toroidal current profile, the safety factor at plasma axis can be set to one by option.

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. The OH flux consumption 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 [10], [11].

In designing the next generation tokamak reactors, the burning time or Ohmic heating (OH) flux supply is sometimes specified. The simplified model is used to estimate how much flux the PF coils can supply to plasma, since the detail calculations such as plasma equilibrium and poloidal field calculation are required to calculate the possible OH flux supply. The infinitive cylindrical solenoid model is used to estimate the amount of the PF coil flux supply. The thickness of the central solenoid coils, Δ_{OH}, is determined so that both current density and tensile stress by hoop force are set to less than the assigned values (input date) : Δ_{OH} is determined as follows.

$$_{OH} = \text{Max.}\ (\Delta_j,\ \Delta_s)$$

Here, Δ_j and Δ_s are the thickness of the solenoid coils required to satisfy the assigned restrictions on the current density and hoop stress, respectively.

$$\Delta_j = B_{ppk}/(\mu_o J_{OH} F_{OH})$$

$$\Delta_s = \frac{B_{ppk}{}^2}{2\mu_o} \Big/ \left(\frac{B_{ppk}{}^2}{2\mu_o} + F_{OH}\sigma_{OH}\right)$$

B_{ppk} : allowable peak poloidal coil field
J_{OH} : allowable solenoid coil current density
σ_{OH} : allowable solenoid coil stress
F_{OH} : occupation factor of OH coil solenoid

The correlation must be obtained between this simple calculation and the detailed one. The correlation coefficient is provided as an input datum in our code. Our code has the function to calculate the possible burn time and/or the suppliable OH flux to plasma by the detailed method at the calculation of time dependent PF coil current patterns. If there is a large difference between the simplified calculation and the detailed one, the correlation coefficient must be modified and calculation must be redone all over again.

In the case of steady state operation by non-inductive current drive, the center solenoid coils can be eliminated by option.

The radial thickness of TF coils, Δ_{TF}, affects the suppliable OH flux and/or the bore radius of the central solenoid as well as other radial build parameters. We have three options to determine the Δ_{TF}.

In the first option, the Δ_{TF} is determined as follows.

$$\Delta_{TF} = \text{Max.} \ (\Delta_j^{TF}, \ \Delta_s^{TF})$$

Here, Δ_j^{TF} and Δ_s^{TF} are the minimum TF coil thickness determined by the restriction of overall current density, j_{tcav}, including TF coil casing and of overall tensile stress, respectively. The analytical solution of TF coil hoop force, F_h^{TF}, is used to calculate Δ_s^{TF}.

$$F_h^{TF} = \frac{\mu_o}{8\pi} I_t \ln(R_o/R_I)$$

Here, I_t is the total current of one TF coil, R_I is the radial position of the inboard TF coil leg, and R_o is that of the outboard TF coil leg. The R_o is determined in the code so as to satisfy two constraints : (1) toroidal field ripple at plasma outer boundary, and (2) radial clearance between shield and TF coil legs.

The toroidal field ripple, $\delta_{rpl}(R)$, is calculated by the following approximate equation [12].

$$\delta_{rpl}(R) = f_{rpl}\left\{\frac{1}{(R/R_I)^{N_{TF}} - 1} + \frac{1}{(R_o/R)^{N_{TF}} - 1}\right\}$$

Here, the f_{rpl} is the correction factor given as an input datum, and the N_{TF} is the number of TF coils. The detailed ripple calculation can be done in the code to check whether the correction factor is adequate or not after all of plasma and machine parameters are determined. If the correction factor is found to be inadequate, the correction factor should be modified and calculation should be redone again.

In both second and third options on the design of TF coil radial thickness, Δ_{TF}, the Δ_{TF} is not determined by overall current density and/or overall hoop stress like the first option, but determined so as to satisfy the conditions that the stress in the TF coil structures such as TF coil casing and the current density in the TF coil conductor are equal to the assigned values. The difference between the second option and the third one comes from the difference of the support system for the TF coil centering force : (1) by center cylinder and (2) by wedge. In the second option, the centering force of TF coils is supported by the center cylinder, and only hoop force is taken into account in calculating the stress in the TF coil structures. On ther other hand, the third option takes into account not only hoop force but also the compressive force in the inboard TF coil casing (inner vault) and the bending force at the corners of inner vault.

The thickness of the center cylinder, Δ_{CY}, is simply determined so that the compressive stress is less than the assigned value. The analytical solution is used for the centering force of the TF coils as shown by the following equation.

$$F_{CNTR} = \frac{2\pi^2}{\mu_o}(R_p B_t)^2 (1 - R_{TF}/\sqrt{R_{TF}^2 - a_{TF}^2})$$

Here, R_{TF} and a_{TF} are major and minor radii of the TF coils, respectively.

The required input power for plasma heating and current drive are calculated after plasma parameters are set. 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.

The resistively and inductively consumed OH fluxes and burn time are also calculated when the options to specify burn time are not used.

Finally, the deposition profile of neutral beam (NB), the amount of shine though and the pressure of energetic NB ions are calculated to check whether the pressure of energetic NB ions given by approximate method is correct of not when the scenario of steady state operation by neutral beam injection is used.

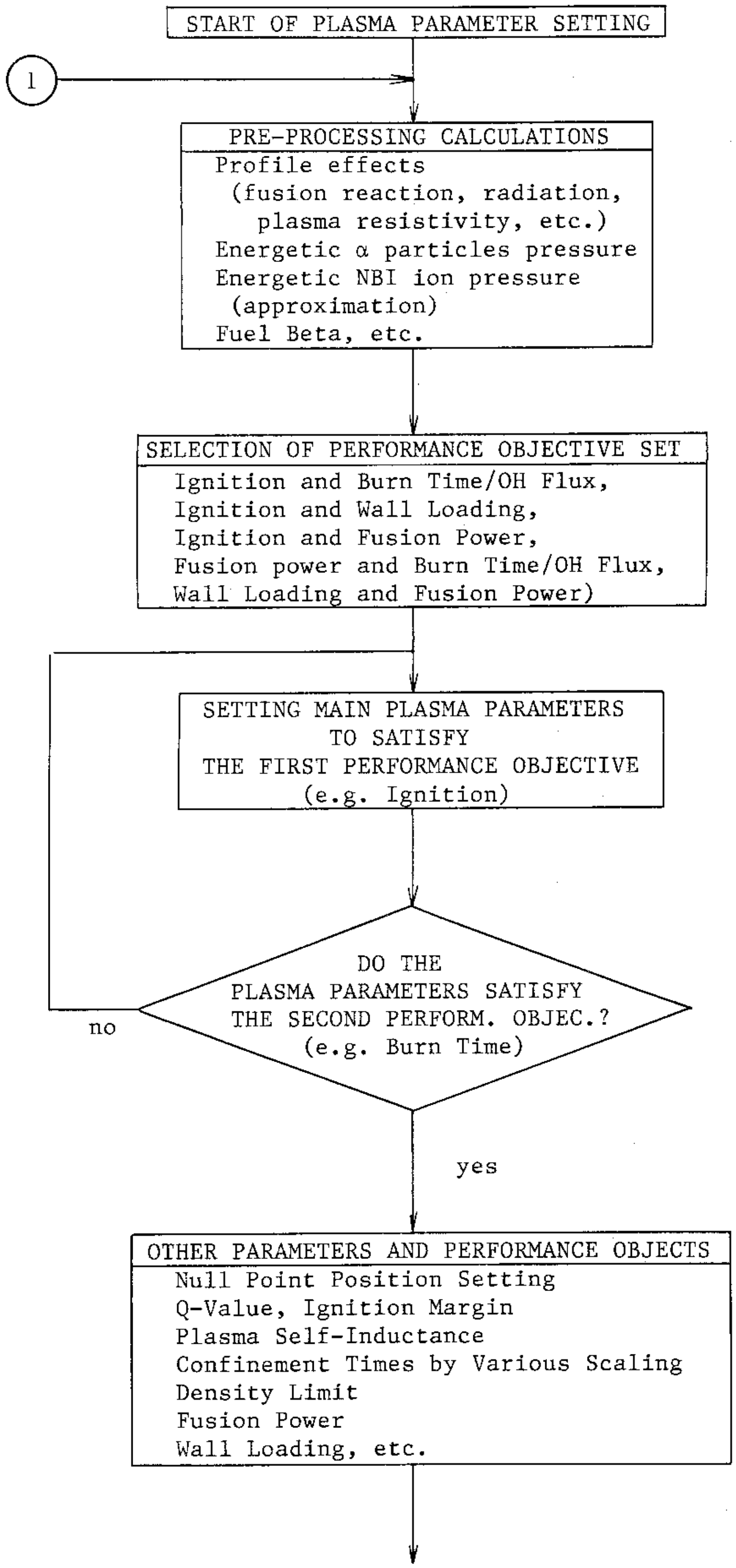
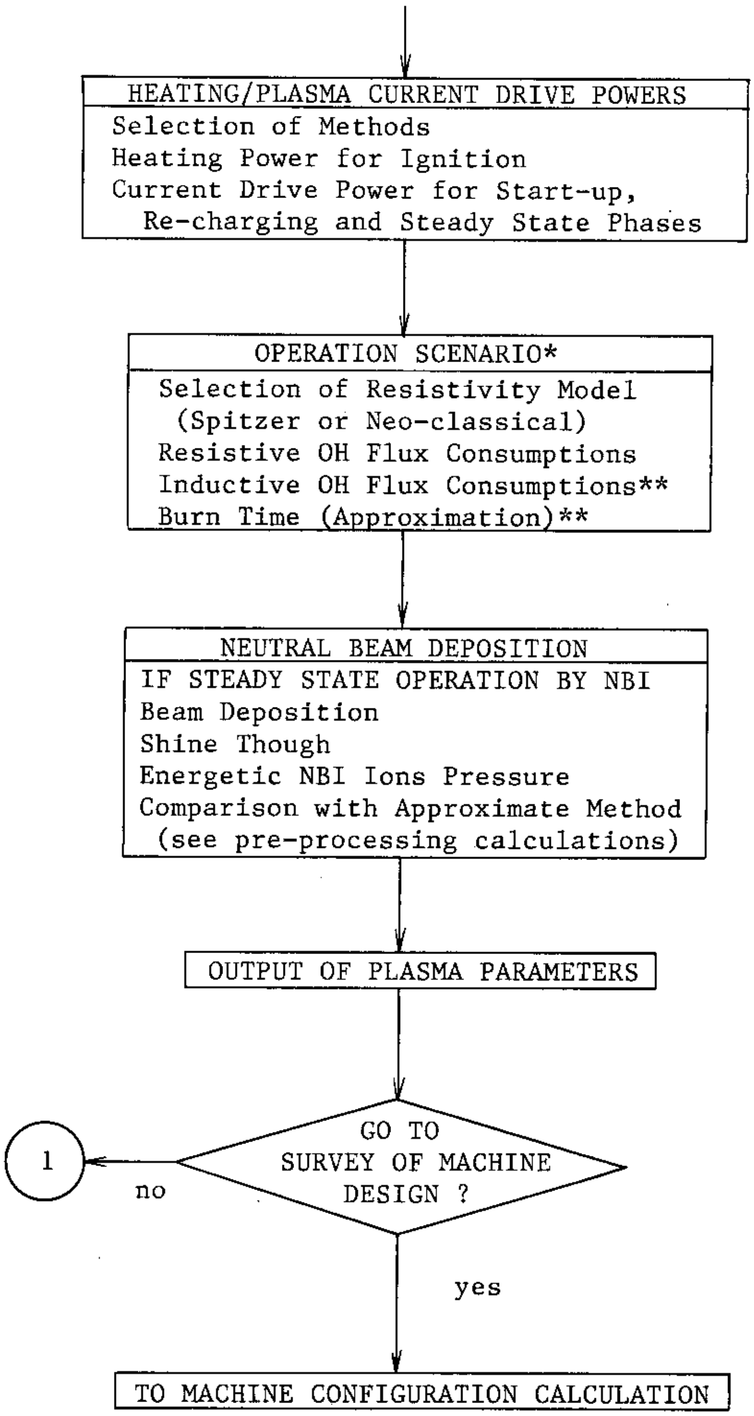


Fig. 2 Schematic Flow Diagram of Plasma Parameter Setting



* When the options to specify burn time, these calculation are already calculated at plasma parameter setting.

** The detail calculation is carried out at determing PF coil currents.

Fig. 2 Schematic Flow Diagram of Plasma Parameter Setting (continued)

2.2 Tokamak system design

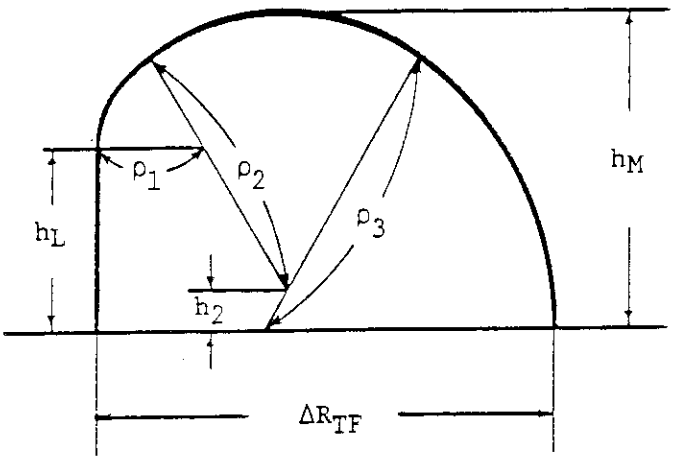
Radial and vertical builds are defined based on the calculations of the plasma parameter setting. Poloidal contour of toroidal field coils is represented by one straight line (wedge part of TF coil) and three arcs in this code. The central angle of these arcs is set to 60 degrees. The radii and the co-ordinates of the arc centers are calculated by the following equations.

$$h_2 = \frac{\sqrt{3}}{2(\sqrt{3}-1)} (2h_M - \frac{1}{\sqrt{3}} h_L - \Delta R_{TF})$$

$$\rho_1 = \frac{2}{\sqrt{3}} h_2 + \rho_2$$

$$\rho_2 = h_M - h_2$$

$$\rho_3 = \frac{2}{\sqrt{3}} (h_1 - h_L) + \rho_2$$



ΔR : diameter of TF coil bore
h_M : TF coil height
ρ_i : radius of the i-th arc
h_2 : height of the second arc center

Poloidal field (PF) coils are located at the effective positions along the line outside the TF coil contour, based on the plasma equilibrium calculations. The prohibited regions against the PF coil arrangement are determined based on the radial and vertical builds. There are three options in determing these prohibited regions, : (1) no restriction, (2) prohibited regions for radial maintenance access and support of torns, and (3) prohibited regions for oblique maintenance access and support of torus. The PF coil currents are calculated, based on the calculations of plasma equilibrium, PF coil current distribution for Ohmic heating (OH) magnetic field, and magnetic fields in the PF coils as described in the section 2.3. Cross-sectional sizes of the PF coils are determined so that the current density in each PF coil does not exceed the maximum allowable value (input data). 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.

The stabilizing property of the torus structures for the plasma vertical instability is estimated in the code by using simplified model, and the feasibility of highly elongated plasma is roughly checked together with the decay index n-value obtained by plasma equilibrium calculations.

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.

The volume and weight of each reactor equipment are calculated, and the weights are summarized in the "Material Table" for each material and each reactor equipment as shown in Table 1. The weight of TF coil shear panels are calculated based on the configuration of TF coils and the prohibited regions against the PF coil arrangement, and summarized in the item of TF coils in the "Material Table". The weight of supper-conducting (SC) coil vacuum chamber is also calculated based on the reactor height and outer radius. The weight of torus spport system is simply defined as some fraction of total tokamak machine weight.

Table 1 Example of Material Table

: REQUIRED MATERIAL TABLE :

UNIT: TON

MATERIAL NAME	DIVERTOR	FIRST WALL	VACUUM VESSEL	BLANKET	SHIELD	TF COIL
SUS 304	0.0	0.0	0.0	0.0	3.2363D+03	0.0
SUS 316	1.7155D+01	0.0	0.0	2.1017D+02	9.2793D+02	2.8695D+03
COPPER	1.9728D+01	0.0	0.0	0.0	0.0	7.4503D+01
LEAD	0.0	0.0	0.0	0.0	1.4509D+02	0.0
BERYLLIUM	0.0	0.0	0.0	1.5095D+02	0.0	0.0
LI2O	0.0	0.0	0.0	5.5048D+01	0.0	0.0
TUNGSTEN	2.3802D+00	0.0	0.0	0.0	0.0	0.0
NB3SN	0.0	0.0	0.0	0.0	0.0	3.4206D+01
GFR	0.0	0.0	0.0	0.0	0.0	1.4302D+01
TOTAL	3.9263D+01	0.0	0.0	4.1616D+02	4.3093D+03	2.9925D+03

MATERIAL NAME	PF COIL	CENTRAL CYLINDER	BASE	BELJAR	TOTAL
SUS 304	0.0	0.0	0.0	0.0	3.2363D+03
SUS 316	9.2563D+02	0.0	1.0009D+03	1.4766D+03	7.4278D+03
COPPER	1.8743D+02	0.0	0.0	0.0	2.8166D+02
LEAD	0.0	0.0	0.0	0.0	1.4509D+02
BERYLLIUM	0.0	0.0	0.0	0.0	1.5095D+02
LI2O	0.0	0.0	0.0	0.0	5.5048D+01
TUNGSTEN	0.0	0.0	0.0	0.0	2.3802D+00
NB3SN	7.8231D+01	0.0	0.0	0.0	1.1244D+02
GFR	4.8306D+01	0.0	0.0	0.0	6.2609D+01
TOTAL	1.2396D+03	0.0	1.0009D+03	1.4766D+03	1.1474D+04

2.3 Magnetic field calculations

The code has two types of magnetic field calculations : (1) one is the calculations necessary to design reactor equipments such as PF coil system, and (2) the other is that to provide only information on design features. As the latter type of calculations, the code numerically calculates the magnetic fields such as toroidal field ripple, peak toroidal field, etc. and the electro-magnetic forces such as over turning force and centering force of TF coils, hoop forces of TF and PF coils, etc. We can skip this kind of calculations to save CPU time by option. As the former type of calculations, plasma equilibrium, current distribution among PF coils for OH field, peak magnetic fields in PF coils are calculated. We can not skip the former type of calculations.

The scenario for time dependent magnetic flux consumption is defined by a plasma current ramp up scerario (input data) and resistively consumed magnetic flux calculated in the step of plasma parameter setting. 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 determined so that the peak magnetic field in all of PF coils are less than the maximum allowable field (input data). The cross-sectional sizes of PF coils are determined after the maximum PF coil currents are determined so that the current densities in the PF coils are limited within the assigned value. since the peak magnetic fields in the PF coils depend on their cross-sectional sizes, this procedure is repeated twice to converge the PF coil sizes.

Through these calculations, a possible burn time and/or an available OH flux supply is obtained. If the burn time and/or available OH flux change is significantly different from the assumed burn time/available OH flux supply calculated in plasma parameter setting, calculation must be redone all over again.

2.4 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 capical costs are calculated by using the characteristic parameters of each system. The cost of land, some buildings such as control room and plasma diagnostics, etc. are assumed to be constant independently on the reactor design. On the other hand, the cost of some buildings such as reactor building, etc. are assumed to be functions of reactor size, thermal or electric output power, neutron wall loading, etc. In the case of reactor power plant designs, the total busbar energy costs are also calculated. The method to evaluate reactor plant economy is basically same as the PNL reports [13], [14].

3. Typical examples of calculations by NEW-TORSAC

In the previous chapter, the structure of the code is described. Here, the capability of the code is described by showing the typical applications. However, it is not so long since the code was developed that all functions of this code have not necessarily been checked enough. So the examples in this chapter are basically come from the INTOR scoping study reported at the previous session [3].

3.1 Automatic design of the next generation tokamak reactor

The NEW-TORSAC automatically designs tokamak reactors from plasma parameter setting to tokamak configuration design, and calculates main characteristic parameters of reactor equipments and auxiliary systems.

Figures 3(a), 3(b) and 3(c) show the typical example of the next generation tokamak reactor designed by this code. The shaded zones in Fig. 3(a) are the PF coil prohibited regions for reactor maintenance access and machine supports. Main characteristic parameters of this reactor are summarized in Table 2. Plasma parameters are determined so as to satisfy two conditions in this example : (1) the ignition condition with margin one, and (2) the assigned burn time ($\sim$1000 sec). ASDEX type τ_E scaling is used to calculate the ignition condition, and Apitzer resistivity is assumed to evaluate burn time. The coefficient of Troyon type critical beta scaling is set to 3.5. The thermal α particle density is assumed to be 5 % of D-T fuel density. The energetic α particle pressure is calculated as $\sim$9 % of fuel pressure which is defined as the summation of D-T fuel pressure and corresponding electron pressure : $(n_{DT} + n_e^{DT})T$, $(n_{DT} = n_e^{DT})$.

Table 2 Main Characteristic Parameters of the Next Generation Tokamak Reactor Designed by NEW-TORSAC.

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)	1.08
Burn time (sec)	1000
Operation scenario	Quasi steady state
Reactor maintenance access	radial access

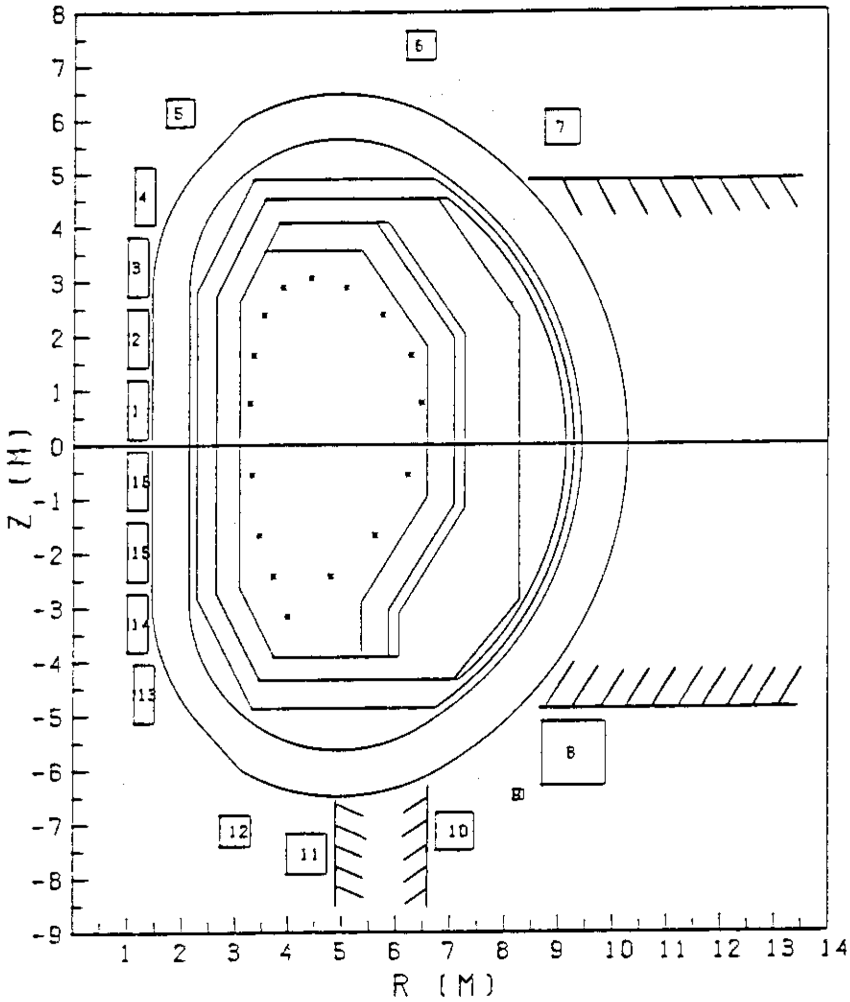


Fig. 3(a) Cross sectional view of the next generation tokamak reactor designed by NEW-TORSAC.

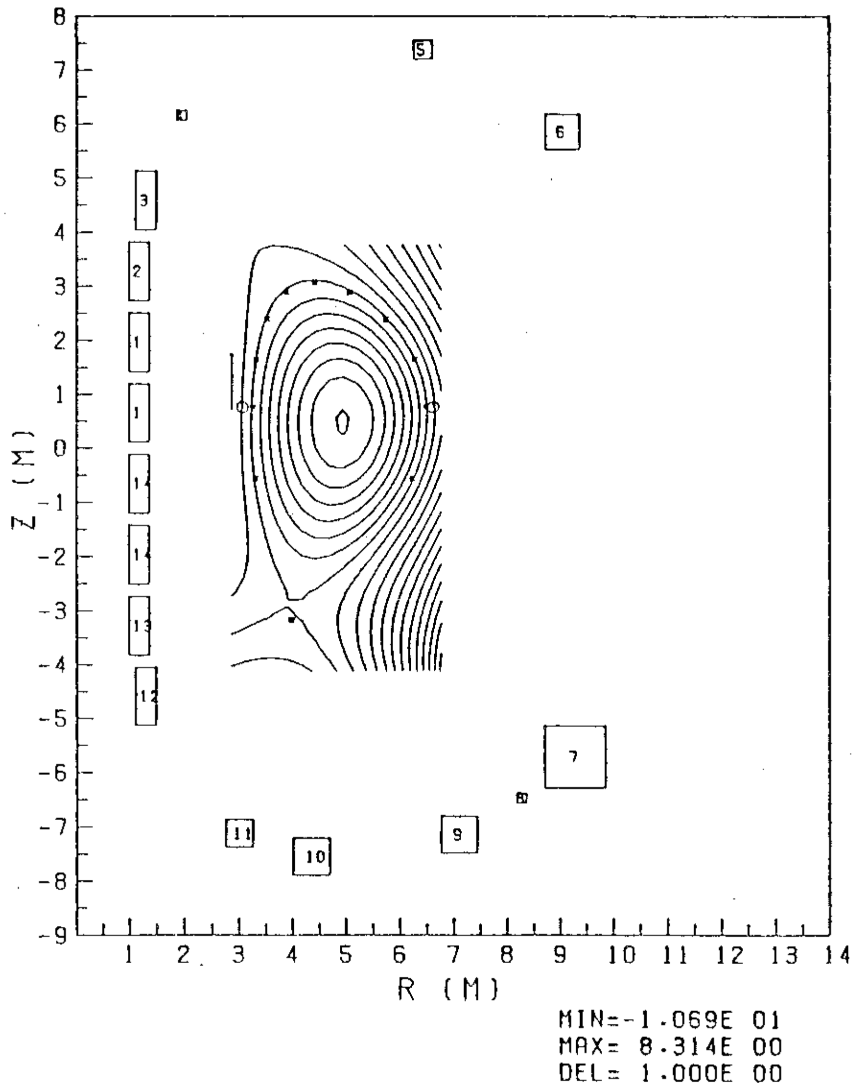


Fig. 3(b) Plasma equilibrium configuration.

high beta phase

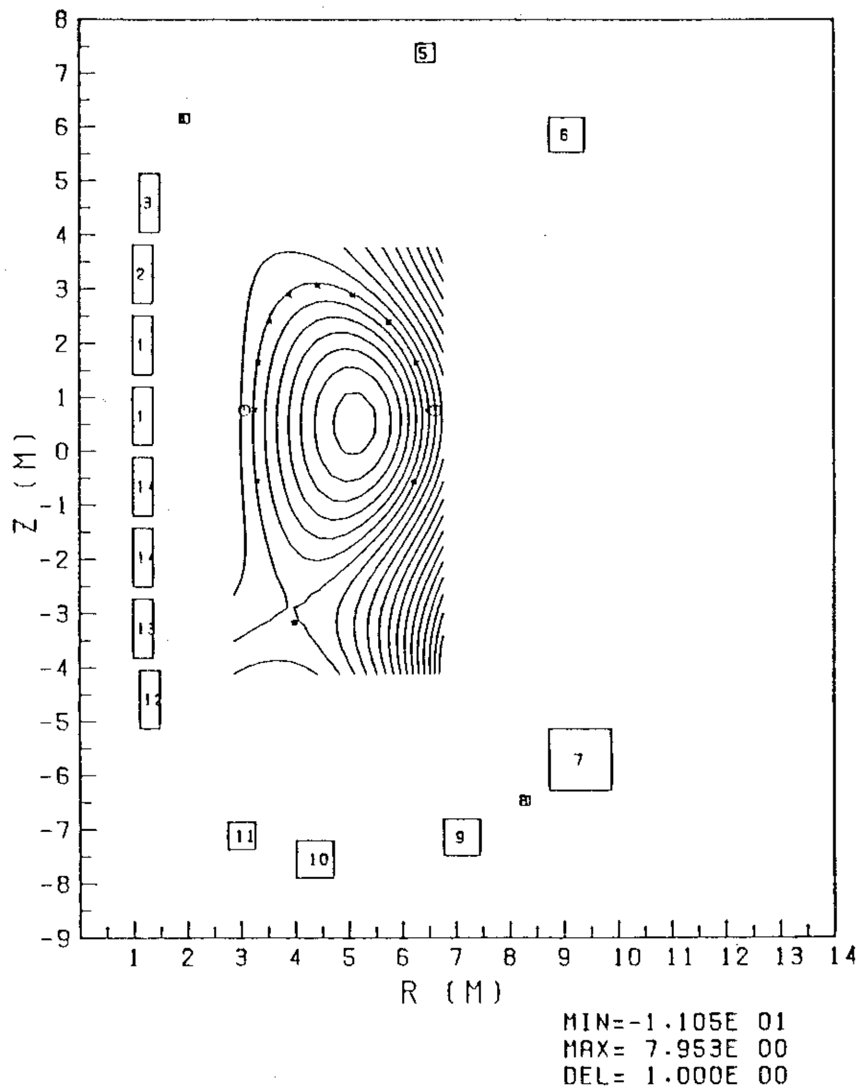


Fig. 3(c) Plasma equilibrium configuration.

low beta phase

3.2 Analysis of tokamak power reactor plant

In designing power reactor plants, the total output power had better be able to be specified. For these situations, the code has options to specify the total fusion power, and neutron wall loading or burn time. Here, we show an example of parametric study on tokamak power reactor plant. The option to specify total fusion power and neutron wall loading is used in this analysis. Figure 4 shows the dependence of plasma size (R_p, a_p) on neutron wall loading. The value of plasma current and toroidal field at plasma center are shown at each calculation point. The totoal fusion power is fixed to 3500 MW. The plasma temperature and the peak toroidal field are 10 keV and 12 T, respectively. The coefficient of Troyon critical beta scaling is set to 3.5. It should be noticed that there is a limit on the obtainable neutron wall loading and there can be two solutions for a given wall loading less than this limit value.

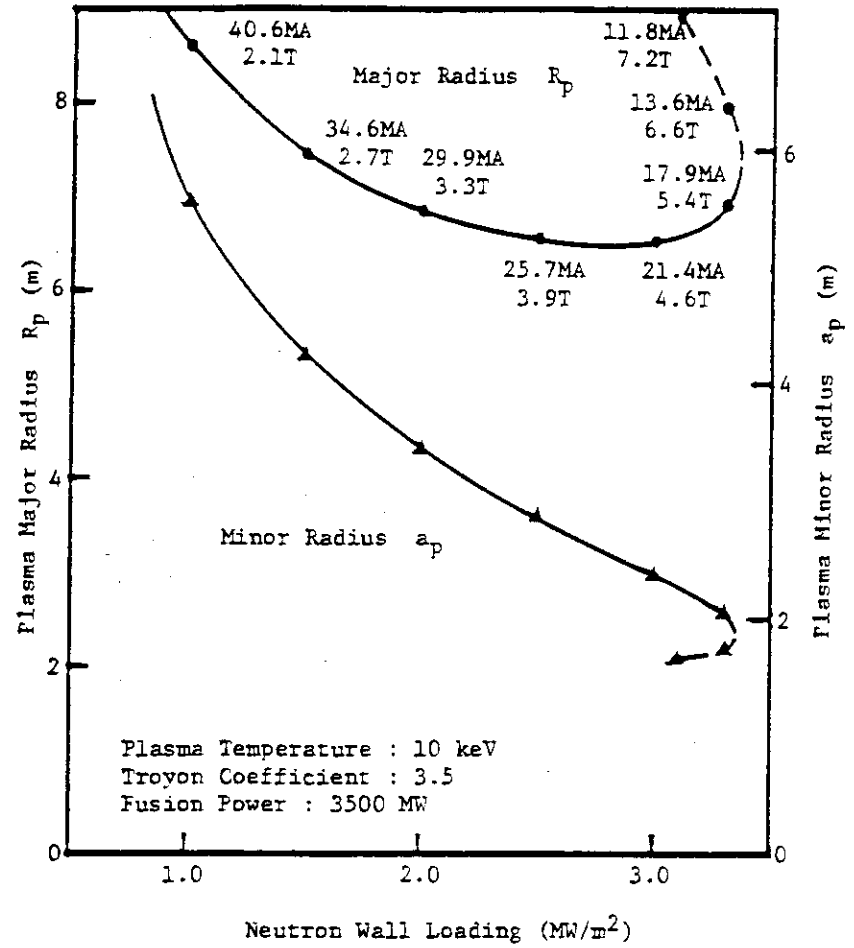


Fig. 4 Dependence of plasma major and minor radii on neutron wall loading.

3.3 Calculation of a possible burn time

Since a possible burn time is automatically obtained through the calculations for determining the time dependent PF coil current patterns, we can survey the dependence of a possible burn time on plasma parameters such as elongation, triangularity, plasma current and so on. As an example of this capability, Figure 5 shows the difference in a possible burn time, $\Delta\tau_{burn}$, between a quasi-steady state operation and an inductive operation as functions of plasma elongation and triangularity. The peak poloidal fields in the PF coils are limited to less than 10 T. The possible burn time of the inductive operation case, τ_{burn}^{Ind}, is fixed 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 result shows that the $\Delta\tau_{burn}$ does not strongly depend on τ_{burn}^{Ind}.

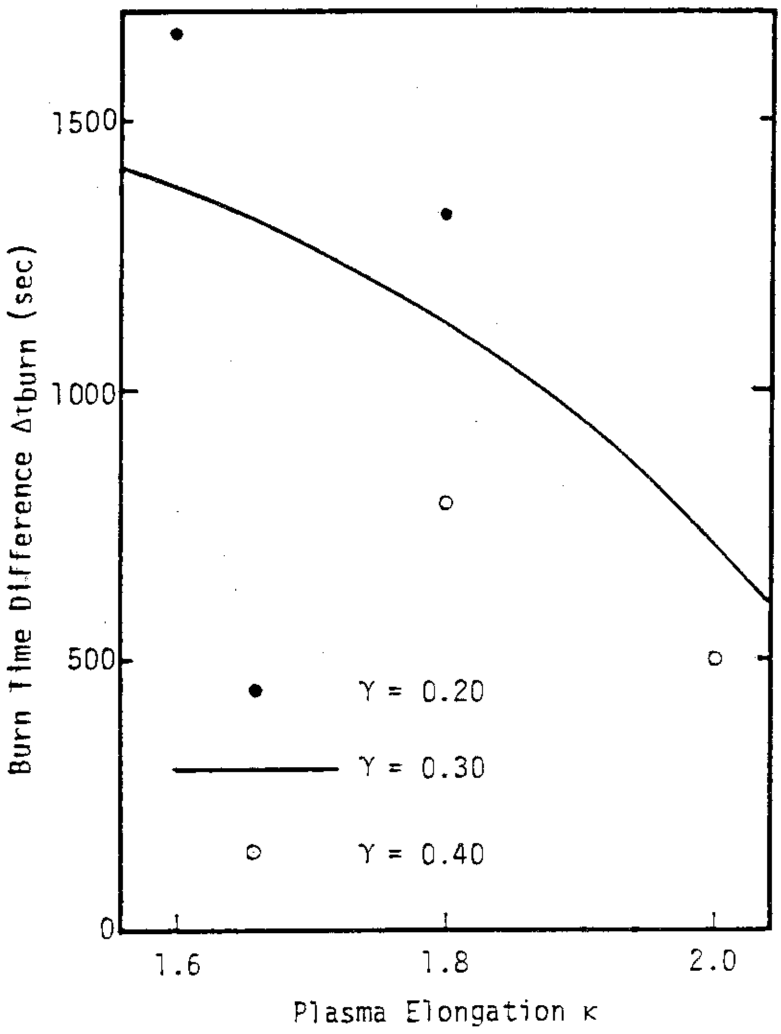


Fig. 5 Difference of possible burn time between quasi steady state operation and inductive operation as functions of plasma elongation and triangularity.

3.4 Restriction on PF coil arrangement for remote maintenance access

As described in chapter 2, the PF coil locations are selected by taking into account both the effective PF coil positions for plasma equilibrium and the prohibited regions against PF coil locations for the remote maintenance access. As an example of this capability, Figure 6 shows the PF coil arrangements of the reactors with radial and oblique access maintenance scenario. Table 3 summarizes main parameters of these two reactors. The possible burn time is ∿1500 sec in both designs.

Table 3 Comparison of Access Directions (κ = 1.8, γ = 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 (MAT)	11.65	7.84
Power Capacity (MVA)	787	675
Total Ampere Turn (MAT)	115	101
Relative Capital Cost	1.0	0.970

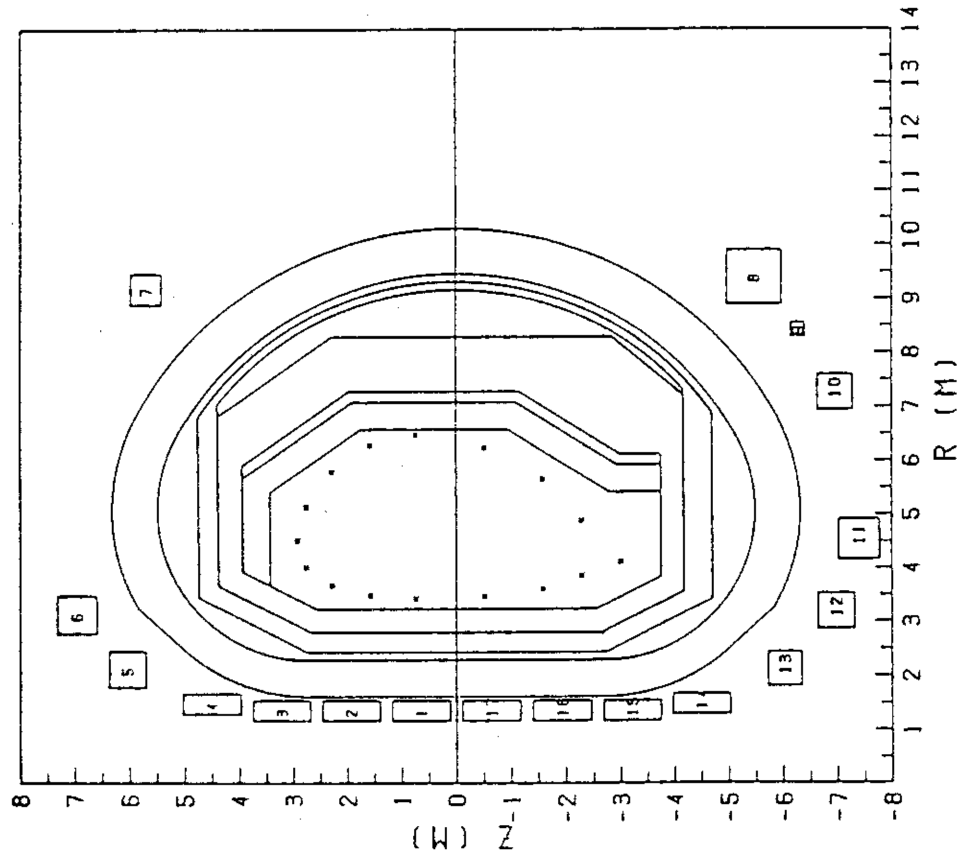


Fig. 6(a) Cross-sectional view of radial access type reactor.

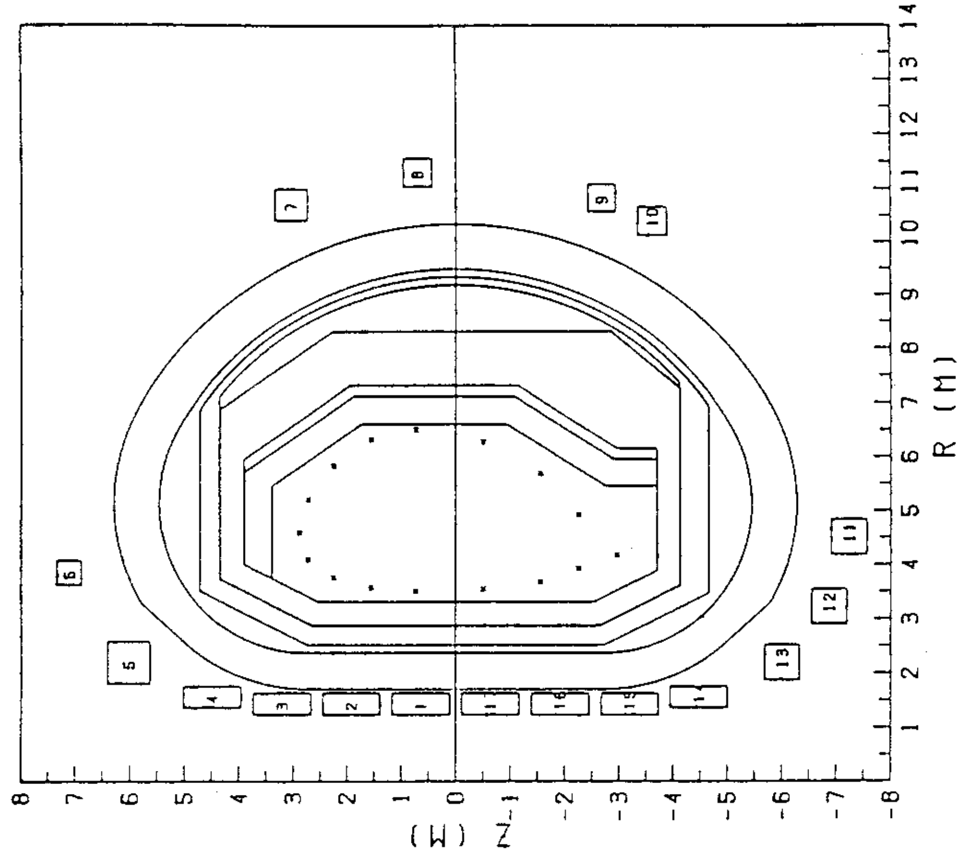


Fig. 6(b) Cross-sectional view of oblique access type reactor.

3.5 Central OH solenoid

In the case of steady state operation by non-inductive method for plasma current drive, the OH solenoid coil is not necessarily required to be installed. For these steady state operation scenarios, the code has an option to eliminate central solenoid coils. Figure 7 shows the cross-sectional view of the reactor without the central solenoid coils and plasma equilibrium configuration at high beta phase. No restriction is imposed on the PF coil arrangement for simplicity. Table 4 summarizes results of comparison study on the methods of full current drive for two PF coil arrangements : (1) with central solenoid coils, and (2) without central solenoid coil.

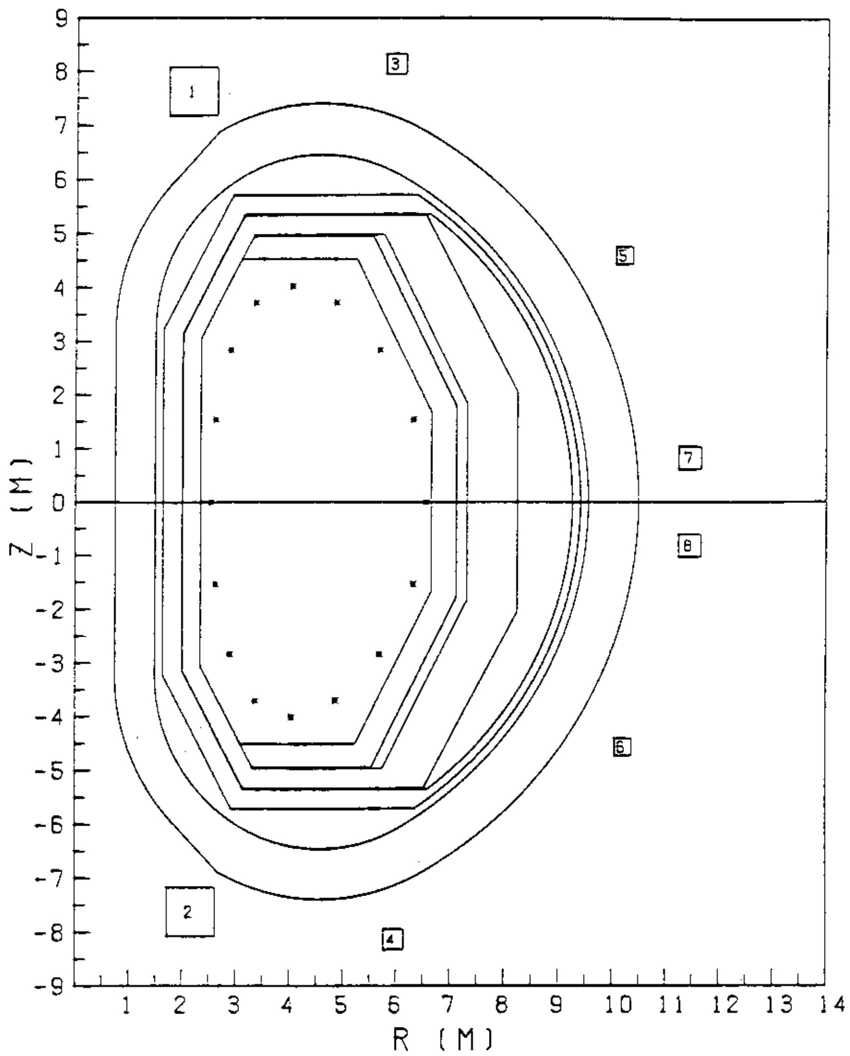


Fig. 7 Cross-sectional view of the reactor without central solenoid coils.

Table 4 Comparison of Steady State Current Drive ($\kappa = 2.0$)

	NBI Steady State Current Drive with Limiter				RF Steady State Current Drive with Divertor	
	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)	4.04	4.55	4.17	4.53	4.13	4.25
Plasma Minor Radius (m)	1.47	2.00	1.25	1.63	1.55	1.32
Plasma Current (MA)	11.3	17.0	9.32	13.7	12.0	9.98
Toroidal Field (T)	3.45	3.06	4.16	3.83	3.37	4.08
Total Weight (ton)	7790	10090	7110	8980	8980	7890
TF Coil Weight (ton)	2200	2850	2110	2560	2340	2240
PF Coil Weight (ton)	740	510	410	500	1070	480
Heating Power (MW)	184	90	185	93	78	80
Wall Loading (MW/m^2)	0.329	0.229	0.384	0.289	0.409	0.477
Peak Stored Energy (GJ)	7.24	6.61	2.39	3.81	8.48	3.66
Power Capacity (MVA)	282	280	47	76	33	104
Total Ampere Turn (MAT)	107	76	83	85	104	83
Relative Capital Cost	0.878	0.883	0.812	0.821	0.888	0.779

3.6 Operation scenarios

Three types of operation scenario can be treated in the code : (1) fully inductive operation, (2) quasi-steady state operation and (3) steady state operation. So the comparison studies can be done among these scenarios. Figure 8(a) and 8(b) show an example of the trade off study between full-inductive operation and quasi-steady state operation scenarios. ASDEX τ_E scaling with ignition margin 2.5 is used in the calculation of Fig. 8(a), and Mirnov type τ_E scaling is used in Fig. 8(b). The dependence of the relative capital cost on plasma major radius is considerably different between these two parametric studies. This difference comes from the dependence of τ_E scaling on plasma size : τ_E of ASDEX scaling depends on major radius, however, that of Mirnov type scaling depends on minor radius.

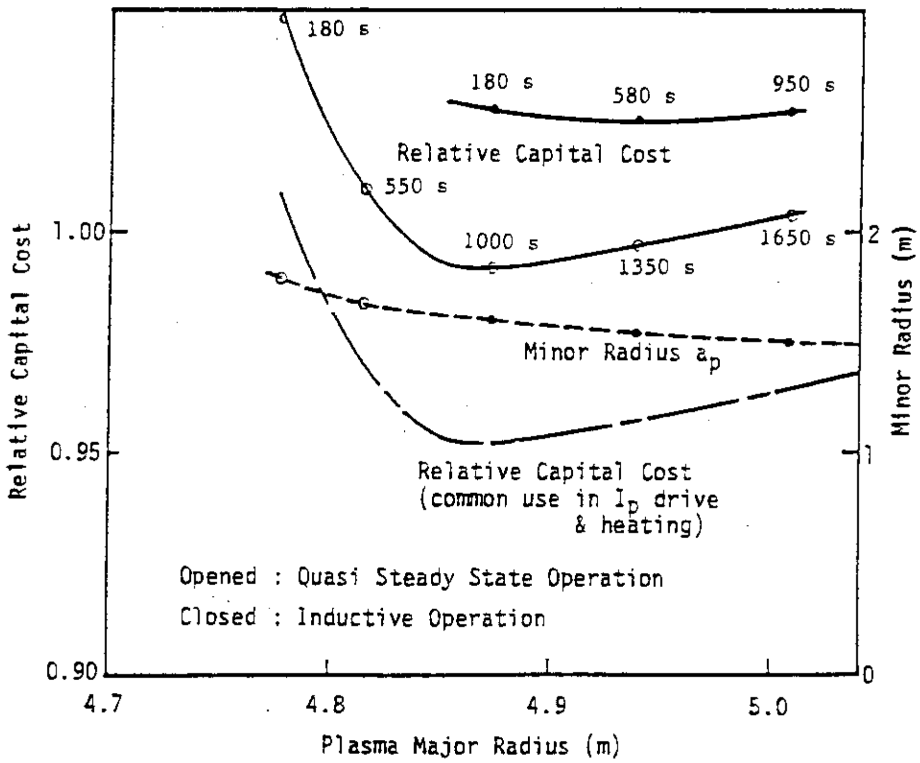


Fig. 8(a) Major radius dependences of the relative capital cost and minor radius for both inductive operation and quasi steady state operation scenario.

ASDEX Type τ_E Scaling

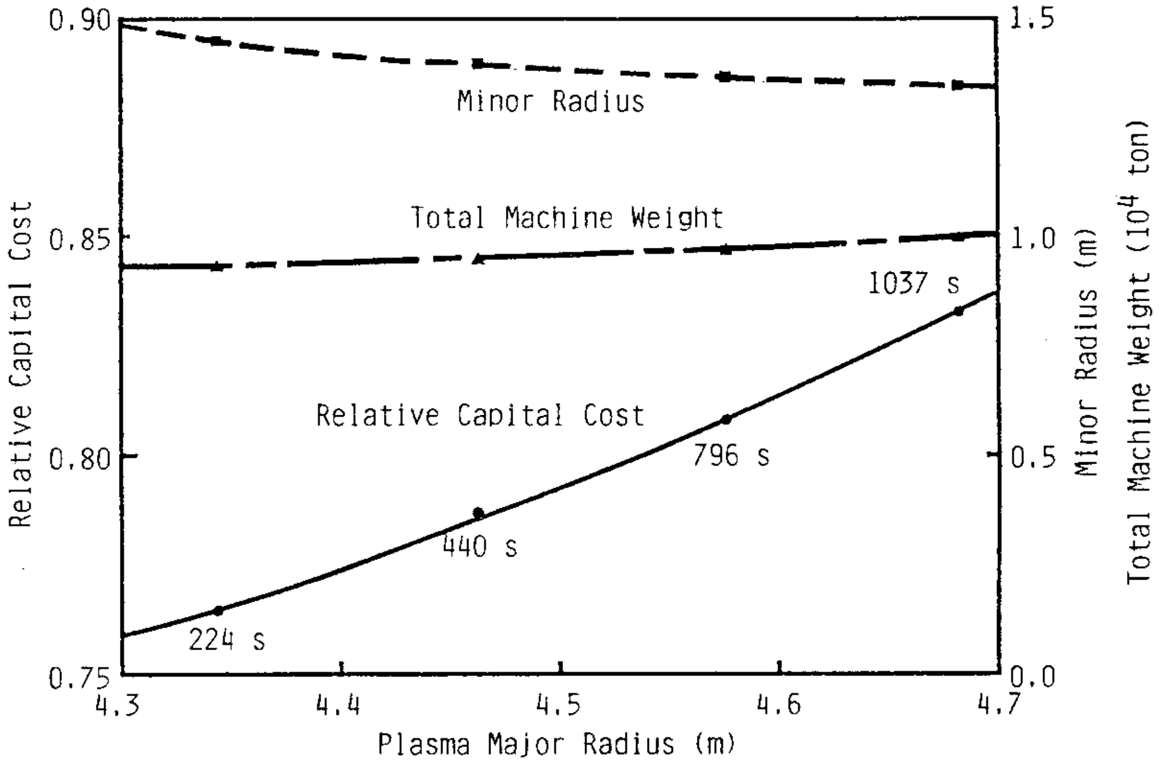


Fig. 8(b) Major radius dependences of the relative capital cost and minor radius for both inductive operation and quasi steady state operation scenario.

Mirnov Type τ_E Scaling

4. Summary

New systems analysis code "NEW-TORSAC" has been developed based on the TORSAC. The NEW-TORSAC is planned to be available for analyzing tokamak reactors from experimental reactors to commercial reactors with supper-conducting coils, and to have a function of automatic design.

Main characteristics of this code are summarized as follows.

(1) The code determines plasma parameters so that the set of plasma parameters provides the desired performance objectives, such as ignition margin, burn time, neutron wall loading, total fusion power, which are assigned by option data. The code can check the performances of the reactor designed by another code or designer.

(2) Though a zero-dimensional model is used in determining plasma parameters, the profile effects are taken into account by specifying the profiles of plasma parameters such as density, temperature, etc. based on 1-D transport simulation and/or experimental results. The profile of plasma toroidal current can be determined so that the safety factor at plasma axis is set to one.

(3) The pressures of the energetic ions (α particles, NBI ions) are calculated in the code by using analytical solution of the Fokker-Plank equation as shown in the reference [7].

(4) The thickness of blanket and/or shield can be consistently determined with the calculated neutron wall loading at plasma parameter setting.

(5) The locations of poloidal field coils are selected at effective places for plasma equilibrium. In determining the poloidal field coil locations, the prohibited regions are defined in the code to ensure the spaces for the torus support structures and the access at remote maintenance.

(6) In the case of steady state operation by non-inductive current drive, the code can design the PF coil system without central solenoid coils and calculate plasma equilibrium under such a PF coil arrangement.

(7) The operation pattern of the poloidal field coil currents are determined by summing up the currents necessary for plasma equilibrium and OH flux supply so that the peak poloidal fields in the poloidal coils are less than the allowable value assigned by input data. Through this calculation process, the burn time and/or available OH flux are also

obtained.

(8) The capacility of plasma vertical position control is roughly evaluated in the code for the designed plasma with assigned elongation.

(9) In order to provide the information on design features, the code has the routines to calculate the magnetic fields (e.g. toroidal field ripple, peak toroidal field, etc.) and the electro-magnetic forces on TF and PF coils (e.g. over turning force, centering force, hoop forces on TF and PF coils, etc.).

(10) Graphical displays are available for many output of this code such as tokamak configuration, plasma equilibrium configuration, electro-magnetic force distributions, etc. as well as numerical output.

(11) Main characteristic parameters are calculated for auxiliary systems such as plasma heating system, coil power supply systems, vacuum system, cryogenic system, cooling system, fuel handling and storage systems, and so on.

(12) Based on the tokamak system design and the characteristic parameters of auxiliary systems, the capital cost of the tokamak reactor is calculated. In the case of analyzing reactor power plant, the busbar energy cost is also calculated.

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