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Comparison between Conceptual design of Main Parameters for Small scale and Large scale TOKAMAKs

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Abstract — *In these paper , discussing about main parameters and design concept of different Tokamaks. In tokamak devices ,having plasma control problem. So that from this tokamaks get the idea about design of small size tokamak like Table-Top Tokamak. The aim of the program is to build the research device to study plasma position, shape and current control problems in the tokamak, as well as to assist the scientific and educational programs on various plasma physics problems.*

Index Terms--*Tokamaks, ITER ,JET , GUTTA tokamak SST-1 etc..*

I.INTRODUCTION

TOKAMAK reactor is large devices which produces the power in MW. In tokamak ,design the different superconducting coil like center solenoid , toroidal field coil , poloidal field coil etc. and design the some parameters based on this coils. A large number of the control field coils allows to control flexibly plasma position and shape. The magnetic field of the central solenoid with air core does not almost penetrate to the plasma location region. Circuits connection of the central solenoid and poloidal field coils allows to control current excepting interference that simplifies a plasma control.

Small tokamaks have played a very important role in fusion research. They have created a scientific basis for the scaling-up to larger tokamaks and established the well known scientific and engineering schools, which are now determining the main directions of fusion science and technology. These tokamak gives idea for build up the small tokamak like table–top tokamak. Researches in controlled thermonuclear synthesis are actively conducted in many countries all over the world. The quantity of working devices and devices under construction and in project is increasing, which causes the necessity in preparing professionals in a wide range of problems in this field.

The teaching and training on the working installation are especially effective. Being simple to operate, flexible to realign and relatively cheap in the sense of the operating cost, tokamak of this type can be used by the universities in the educational process in the frame of joint educational and scientific projects with other universities and research laboratories all over the world.

TOKAMAK

The tokamak was first developed in the Soviet Union in the early 1960s, the name “tokamak” being formed from the Russian words toroidalnaya kamera and magnitnaya katushka meaning “toroidal chamber” and “magnetic coil”The advantage of toroidal, as opposed to linear, geometry is obvious from the avoidance of “ends”. The simplest magnetic field in a torus would be a purely toroidal magnetic field in which all the field lines form circles passing round the torus. However, a plasma placed in such a field cannot come to an equilibrium force balance. The pressure of a toroidal plasma would cause it to expand and the toroidal magnetic field is unable to provide a balancing force.[3]

In the tokamak this difficulty is overcome by passing a toroidal current through the plasma itself. This current produces a poloidal magnetic field, whose field lines pass the “short way” round the plasma. This encircling magnetic field is able to hold the plasma in place and to provide an equilibrium force balance. The way in which the toroidal and poloidal magnetic fields combine is illustrated the resulting field lines taking a helical path around the torus .

In order for the fuel in the form of plasma to produce enough thermonuclear reaction , it must be maintained in a limited volume and kept away from any structural material in order to maintain its high temperature .This is called confinement .

As the plasma is made up of charged particles, the magnetic fields may act on them. If this same plasma is bathed in a rectilinear magnetic field , the particles wind around the field lines and will no longer touch the side walls .

So as to avoid losses from the edges, we close off the magnetic bottle by creating a torus. The magnetic field thus created by a series of magnets surrounding the plasma is called a toroidal magnetic field. The magnets generating this field are the toroidal magnets. Here is shown that confinement is not quite enough and to minimize particle leakage even more, the field lines must be helicoidally. This is achieved by adding another magnetic field to the toroidal field, which is perpendicular to it (the poloidal field).The equilibrium of the plasma, its position, its shape and the control of the current are taken care of by the group of horizontal magnets called poloidal coils.

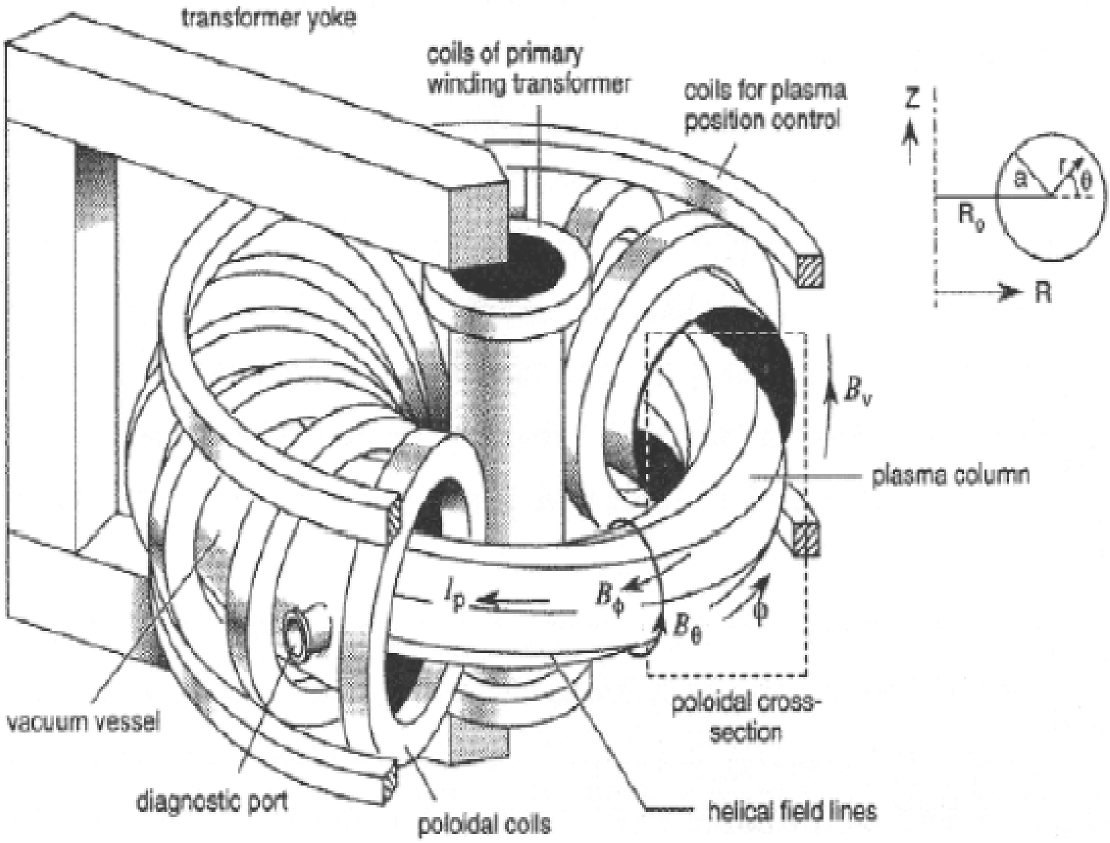

Fig. 1. ***TOAKAMAK***

II. MAIN COMPONENT S OF TOKAMAKS

a) Plasma

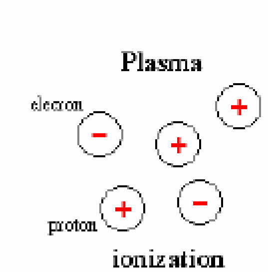

Fig. 2. ***Plasma: The fourth states of matter***

There are three classic states of matter: solid, liquid, and gas; however, plasma is considered by some scientists to be the fourth state of matter. A plasma is a gas in which charged particles are of sufficient importance for the gas to be a good electrical conductor. Ordinary matter becomes ionized and forms a plasma at temperatures above about 5000 K, and most of the visible matter in the universe is in the plasma state. The high electrical conductivity implies that currents can flow in a plasma. These currents can interact with magnetic fields to produce the forces that are needed for confinement.[2]

b) Vacuum Vessel

The basic purpose of the vacuum vessel was to hold a vacuum in which the pressure was less than one millionth of atmospheric pressure. This meant of course that it would have to carry the force of atmospheric pressure over the whole of its surface, 10 tones per square meter over an area of 200 square meters.

In order to cleanse the plasma-facing surface of the vessel of impurities it was designed to be baked at 500oC, and this implied the additional requirement that the heating and cooling had to be carried out without unacceptable stresses from expansion and contraction. The vessel was designed with a double skin to allow heating by hot gas which is passed through the interspaces.

The material for the vacuum vessel should be non magnetic so as not to shield or distort the externally applied magnetic fields. The resistivity should be such that the eddy currents generated by time varying magnetic fields would dissipates in times far smaller than the characteristics rise times.

c) Magnetic Field Coils

A Tokamak type of magnetic confinement device employs three principal sets of magnetic field producing coils .These coil sets produce:

a) The main toroidal field (TF) Which confines the plasma ;
b) The ohmic heating (OH) power and the transformer flux ;
c) The vertical or equilibrium or Poloidal field(PF) that keeps the plasma in an equilibrium position .

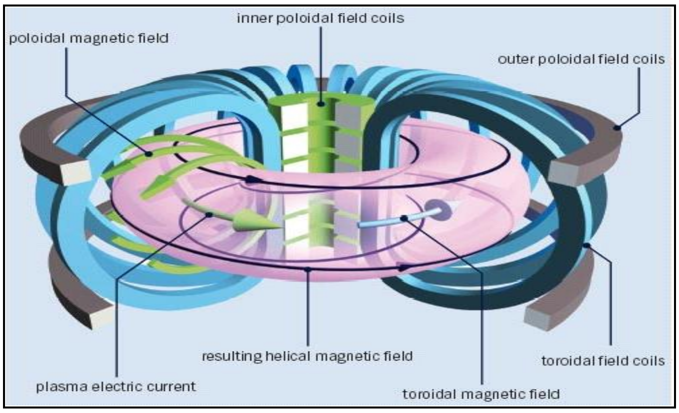

Fig. 3. Magnetics field coils

The toroidal magnetic fieldconfine the plasma in proper shape. It is Provided on the vacuum vessel . Each coil has wound with no turn of conductor. The coils are to carry currents for several tens of seconds, and consequently they had to be provided with a cooling system, using water as the coolant. The toroidal magnetic field is to be produced by a finite number of coils equidistant from the major axis of the torous and lying in planes passing through the major axis .

The center solenoid coil has group of 6 to 8 coils composes the primary act of the transformer, which allows to heat, the plasma in its initial phase. The primary purpose of the ohmic transformer is to initiate plasma discharge and drive plasma current which establishes a poloidal field as well as heats the plasma ohmically. Plasma current forms a closed loop and it must be driven inductively by an external transformer. The plasma loop thus forms a single turn secondary of transformer in which the transformer primary must induce a voltage to drive ,heat and sustain plasma current .

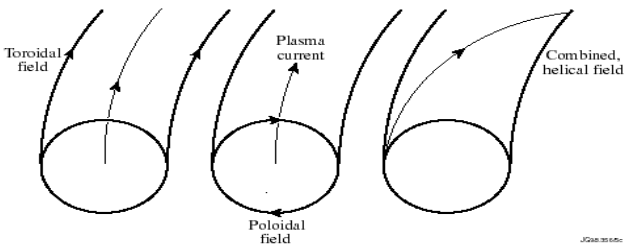

Fig. 4. ***The tokamak's toroidal magnetic field combines with the poloidal field of the plasma current to produce a magnetic field with helical field lines.***

The poloidal field coils are horizontal circular coils. If these coils were placed inside the toroidal field coils the two sets of coils would be linked, with the associated problems of assembly. The poloidal field coils are therefore placed outside the toroidal field coils. The main poloidal field coil is the inner coil wound round the central column of an iron transformer core, to act as the primary of the transformer. The other six coils are optimally placed to provide control of the plasma shape and position.

d) Power Supplies

Electrical power was needed to supply the currents in ohmic transformer coils, both the toroidal and poloidal field coils, a similar power being required for each.

Tokamaks are designed to allow a pulse repetition rate of one every some minutes. Each pulse would call for a total power of up to 100- 1000MW - the output of a medium sized power station. In previous machines this pulsed load had been dealt with using heavy flywheels driven up to speed by motor generators, the energy in the flywheel then being extracted during the plasma pulse. The system used for Tokamak combined this procedure with several hundred megawatts taken directly from the electricity grid.

e) Plasma Heating Systems

It is decided that the main heating would be provided by two heating systems - neutral beam injection and ion cyclotron resonance heating. Each of these systems consisted of a number of separate units, so the power could be increased over time by increasing the number of injectors and antennae. The plan was for an initial installation of a few megawatts in each system, to be increased later to 25MW of neutral beams and an ion cyclotron power of around 15MW.The following name of some methods for heat the plasma :

1. OHMIC HEATING
2. NEUTRAL BEAM HEATING
3. RADIO FREQUENCY HEATING

III. DEFINE THE PLASMA PHYSICS PARAMETERS

a) Safety Factor

In a Tokamak device, the relationship between the toroidal current and the edge safety factor is very important because this will determine the eventual device size according to MHD stability requirements. The safety factor is given by,

$$q = \frac{5a^2 Bt}{R * Ip}$$

In this expression, the toroidal magnetic field Bt is measured in Tesla, the minor radius a and the major radius R in meter, the toroidal current I p in MA. Safety factor has to be assumed. Its value is taken between (2.5 to 5).

For smaller safety factor less than 2.the current value can be increased without changing field or minor radius value. But that leads to increased plasma disruptively and high energy losses. Improvement in confinement with increased plasma current saturates at q=2.5.Plasma is more stable at q>2.5 and reliable with respect to kinks.

The edge safety factor at the 95% of the flux surface gives the range of field and current. The safety factor at the centre and at the edge of the plasma defines the stability of the plasma. The safety factor at 95% of flux surface can be given by,

$$q_{edge} = \frac{q*(1.17-0.65*E)}{(1-E^2)^2}$$

Where, E= inverse aspect ratio

b) Aspect ratio

The main machine parameters and the cost change with increasing aspect ratio in following way.The toroidal field, the magnetic energy increase with aspect ratio. The minor radius and the plasma current decrease with aspect ratio.

However the cost of the machine stays constant over most of Aspect ratio range investigated. For Tokamak aspect ratio should be 2 to 5.

Aspect ratio, A=R/a

c) Density limit

For the density limit, the Murakami – Hugil limit which is closely related to radiation cooling in Tokamak fuelled by gas puffing is adopted:

$$n \underset{-}{<} n_{MH} = \frac{2B_0}{R_0 q^*}, \left(10^{20} m^{-3}\right)$$

d) The effective ion charge, Zeff

The effective ion charge, Zeff represents the average molecule mass Z i of gases inside the system, which indicates the level of the impurities in the plasma. The impurities can be chemical elements alone or a wide variety of compounds coming from the walls of the vacuum vessel and from the peripheral systems of the Tokamak. As the impurities have Z i >> 1, in order to approximate Zeff to the Z of the working gas, the Tokamak chambers have to be adequately conditioned with the aim of avoiding the sputtering of particles in the plasma-wall interaction process.

e) Plasma Temperature

The electron temperature is calculated according to the energy balance equation for a Tokamak discharge.

Electron temperature

$$Te = \frac{I_p V_l \tau_E}{2 * k * \pi^2 R * r^2 n_e} \quad \text{ev}$$

Where, Ip is the plasma current in A, Vl is the loop voltage in V, k is the Boltzman constant, R and r are the major and plasma radii in m.

Ion Temperature

$$T_i = 0.032 * \sqrt[3]{(I_p * B_t * R^2 * n)} * \sqrt{A} \text{ ev}$$

f) Confinement time

In Tokamak confinement time is given by some empirical scaling law. Using the neo-AL-CATOR scaling law valid for ohmic heating of plasma which is not too dense, the energy confinement time is given as

$$t_s = 7 * 10^{-22} \eta q a_p R^2 \text{ Sec}$$

This energy confinement time is used for small Tokamak. As confinement time is increased size of tokamak increased.

IV. DETAIL OF DIFFERENT TOKAMAKS

In the world ,many large tokamaks are build up in many country Research is going on for making good successfully tokamak. Now at present , ITER tokamak is very huge tokamak . The ITER machine, thus developed represents a step which will demonstrate the plasma physics as well as much of the technology of an electricity producing fusion power plant. In doing so, it will bring the world to the threshold of practical fusion energy.[4]

A. Large scale tokamak database.

Some detail of some different large scale tokamaks :

TABLE I. LARGE SCALE TOKAMAKS

Tokamak	***R,m***	***a,m***	***I_p,mA***	***b/a***	***R/a***	***B_t,T***
ITER	6.0	15	2	2	2.79	4.85
JET	3	1.25	4	1.6	>2.4	3.6
START	0.32	0.26	0.260	<3	>1.25	0.32
MAST	0.7	0.5	1-2	<2.5	>1.3	0.63
SSTR	7	1.75	12	1.8	4	9
JT-60U	3.1	0.9	4	1.7	3.4	4
JT60SC	2.8	0.9	3	1.9	3.1	3.8

SST-1	1.1	0.2	0.220	1.7-2	5.2	3
K-STAR	1.8	0.5	2	2.0	3.6	3.5

R-Major Radius , a-minor Radius , I_p–plasma current. b/a- elongation, Aspect ratio-R/a, Bt- Toroidal magnetic field
International Thermonuclear Experimental Reactor (ITER)[5],
Joint European torus (JET)[3]
Small Tight Aspect Ratio Tokamak (START)[8]
Mega Amp Spherical tokamak (MAST)[9]
Steady state Superconducting tokamak (SST-1)[1]

B. Small tokamak database.

As one of the first CRP (Co-ordinated Research Project) activities, the small tokamak database has been established, Table2. There were several attempts in the past to create a small tokamak database.[10]These are creating a new updated database with 10 presently or recently active small tokamaks and devices under construction in it. The selection criteria for the database was chosen not exclusively on the device size, but on the scale of a project in whole, including budget and staff consideration. For example, the small-size Alcator C-Mod was not included in this draft version, but the ET tokamak with 5m major radius and the medium size T-10, HL-2A and HT-7 are included. In other words, we include tokamaks that could more efficiently benefit from the CRP network. However, any small and medium size tokamaks are welcome to join the CRP and to contribute to its activities. This table is far from being complete and only includes information available to the authors. It is one of the objectives of the CRP to complete it and update on a regular basis. Numbers after a slash in the Table give design values if they have not been yet achieved. Blue colour represents closed projects, pink – projects under construction, for devices marked in black information is provisional. [6]

TABLE II. SMALL SCALE TOKAMAKS

Tokamak	***R,m***	***a,m***	***I_p,mA***	***b/a***	***R/a***	***B_t,T***
Globus-M St.Petersburg	0.36	0.24	350/ 500	1.6 /2.2	1.5	0.5/ 0.62
GUTTA St Petersburg	0.16	0.08	150	2	2	1
CASTOR Czech	0.4	0.085	20	1.0	4.70	1.5
ISTTOK Portugal	0.46	0.085	12	1.0	5.41	0.5
Compass-D UK	0.557	0.232	350/ 450	1.66	2.4	2.1
TRIAM JAPAN	0.8	0.12	--	1.8	6.6	8
KT-1 KOREA	0.27	0.05	15	1.0	5.4	1.5
HT-7 CHINA	1.22	0.35	400	1.0	3.48	3
ADITYA INDIA	0.75	0.25	30	1.0	3	3
SINP INDIA	0.3	0.075	75	--	4	2

V. CONCLUSION

From study of different tokamak make the conceptual design of tokamak. Furthermore, small tokamaks are very convenient to develop and test new materials and technologies, which because of the risky nature cannot be done in large machines without preliminary studies. Small tokamaks are suitable and important for broad international cooperation, providing the necessary environment and manpower to conduct dedicated joint research programmed. In addition, the experimental work on small tokamaks is very appropriate for the education of students, scientific activities of post-graduate students and for the training of personnel for large tokamaks. All these tasks are well recognized and reflected in ITER documents and understood by the large tokamak teams.

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