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FEBRUARY 1980 PPPL-1632

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OCLATOR
(One Coil Low Aspect Toroidal Reactor)

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OCLATOR

— One Coil Low Aspect Toroidal Reactor —

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ABSTRACT

A new approach to construct a tokamak-type reactor(s) is presented here. Basically the return conductors of toroidal field coils are eliminated and the toroidal field coil is replaced by one single large coil, around which there will be placed several tokamaks or other toroidal devices. The elimination of return conductors should, in addition to other advantages, improve the accessibility and maintainability of the tokamaks and offer a possible alternative to the search for special materials to withstand large neutron wall loading, as the frequency of changeover would be increased due to minimum downtime.

It also makes it possible to have a low aspect ratio tokamak which should improve the β limit, so that a low toroidal magnetic field strength might be acceptable, meaning that the NbTi superconducting wire could be used. This system is named OCLATOR (One Coil Low Aspect Toroidal Reactor).

OCLATOR

— One Coil Low Aspect Toroidal Reactor —

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1. Introduction

The engineering difficulty of designing a tokamak resides in improving accessibility without increasing the cost excessively. The author has proposed previously a lower number coil design as a possible alternative approach to design an ETF device.[1] Reducing the number of coils improves the accessibility, but the ripple considerations[2] dictate a larger diameter for these coils and hence increase the cost. By looking at any conventional tokamak design, one notices that the return legs of the toroidal coils do not serve any purpose (except obviously satisfying the condition div. $\vec{J} = 0$), but do cause a lot of difficulties. It looks, at first glance, impossible to eliminate the return legs. But it can be done. The concept is using a gigantic one looped coil and to string many tourses around the one coil. The diameter of the TF coil could be 150 to 200 m. The number of toruses mounted on the loop could be 10 to 15. (See Fig. 1). The uniformity of the field encircling the coil can be insured by applying a vertical field to the plane of the TF coil and possibly providing quadrupole fields either by coils or by magnetic iron material which could easily double as either blankets or walls. The number of toruses and the diameter, of course, should be determined by a more detailed cost analysis. It is suffice to say that each torus may need an equivalent of 40 m ~ 50 m length of the TF coil. This length is comparable with the total length of the TF coils in a conventional tokamak design. The drawback is, of course, this system is worthwhile only when several toruses are operational in one location. We shall see later, though, this may yet have its own merit.

In what follows, we shall describe (1) design of a tokamak type reactor, (2) construction consideration of the TF coil and auxiliary coils, (3) incorporation of new ideas for the tokamak and (4) the system consideration of this one coil, low aspect toroidal reactor (OCLATOR).

(1) Design of tokamak-type reactor.

Once the single coil idea is introduced, we can construct a low aspect ratio tokamak. Usual low aspect ratio tokamak could be constructed in a conventional coil system, but the removal of the return legs seems to reduce the engineering constraint enormously and the startup of the plasma current could also be improved.

(1a) Design parameters and finite β limitations.

The finite β limit tends to limit the $n\tau$ requirement. Let us assume that the Alcator scaling law applies. Then

$$n\tau T^2 \propto \beta^2 B^4 a^2$$

here we use the suitably averaged quantity for β (i.e. $\beta = \bar{\beta}$).

Thus improving β is obviously a very good approach to reduce B or a, hence cost. The β limit, β_c in the conventional numerical analysis approach indicates that

$$\beta_c = \frac{a}{R} \frac{1}{q^2} f(\kappa) \qquad (1)$$

where R/a is the aspect ratio, q safety factor at the plasma edge, $f(\kappa)$ is a function of elongation parameter $\kappa (1 \lesssim f \lesssim 2)$. The above ballooning stability limit is, however, dictated by the presence of shear. Hence q cannot be too small. Typically $q \simeq 2.5$. On the other hand, if the magnetic well stabilization is considered, as is well-known,[3] the critical β may be given by

$$\beta_c = \frac{a^2}{R^2} \frac{1}{q^2} g(\delta) \qquad (2)$$

where δ is a shaping factor (such as triangulations) and $g(\delta)$ is between 1 and 2 (possibly more). At first sight, β limit of Eq. (2) seems inferior to Eq. (1). However, because shear is not a dominant factor for stability in Eq. (2), q could be lowered. In fact, there are experimental results indicating that (edge) q could be lower than 2. Hence if we let q = 1.4 (still maintaining q_o=1), in principle, at R/a = 3.0, Eq. (2) has higher β than Eq. (1). In practice, of course, more careful study (both theoretical and experimental) is needed to give definite comparison, however, I think, there is a prima facie case for improved β in the lower aspect ratio.

The design which gives the same $n\tau$ as INTOR/ETF parameters but assuming $g(\delta) = 1$, $q = 1.2$ and 1.4 using Eq. (2) is shown below.

q	a(m)	Δ(m)	R(m)	B at plasma (T)	β(%)	I(MA)	B_{max} at coil (T)
1.2	2	1.5	6.07	3.74	7.50	10.27	8.83
1.4	2	1.5	6.07	4.36	5.51	10.27	10.30
1.2	2	1.3	5.76	3.55	8.38	10.27	8.3
1.4	2	1.3	5.76	4.14	6.16	10.27	9.68
1.2	2	1.0	5.29	3.26	9.94	10.27	7.53
1.4	2	1.0	5.29	3.80	7.30	10.27	8.78

The above table is composed with the assumption that the plasma volume be minimum without making B_{max} excessively high. Here Δ is the total distance between the surface of the superconducting wire and the plasma edge. For INTOR/ETF, this Δ is about 1.3 to 1.5 m. This distance, Δ, in the table may look small, but the argument given below in Section (2) may make this distance practical. In any case if $g(\delta)$ could be made greater than 1.2, there will be further advantage for this design.

(b) <u>The start-up for plasma current.</u>

One problem of the low aspect ratio torus (see e.g. SMARTER)[4] is how to start ohmic heating current. In this configuration of one coil, there is enough space available for both compression and vertical translation. (In a conventional sense, i.e. parallel to the major axis.) The compression can be used to generate ohmic current by using the space to be occupied by the compressed plasma. Similarly the vertical translation could be used to provide the necessary volt-seconds by using the area between the superconducting coil and the plasma edge. The staging area need not be heavily shielded: thus the area could be used to place the primary winding and preliminary heating equipments.

(1c) Ripple consideration.

One single coil, even 150 to 200 m diameter with the minor radius of 2.2 ~ 2.6 m could still experience the field ripple. If the ripple workshop report is to be believed, however, a single perturbation (such as due to the bundle divertor) appears to be less serious than multiple ripples. In this report, we will not go into detail, but it appears that the first moment, i.e. dipole moment could be conveniently canceled by applying a uniform field perpendicular to the plane of the coil whose strength is approximately equal to $\alpha B_{max}/A_c$. Here α is a number of the order of unity, A_c is the aspect ratio of the single coil ($29 \leq A_c \leq 44$). Thus several kilogauss of the uniform field will be more than sufficient. Incident.lly this field to the first order, will make the single coil force-free.

The remaining quadrupole moment is probably tolerable, but, if necessary, a judicial choice of magnetic material for the building wall and/or the shaping of the minor cross section of the single coil would eliminate this.

The curvature of the single coil may be eliminated by making the single coil into a polygon shape (say 20). It appears though, the distortion which amounts at most to 8.3 cm at 5 m away from the plane of minor axis, could be tolerated.

There are other considerations for facilitating the design of tokamak plasmas. However, they will be discussed later.

2. Engineering design considerations.

A detailed analysis is needed to look into the feasibility of a gigantic single coil. However, several factors may make this design not too impossible. (sic.) The foremost advantage is, of course, the coil is circular. Hence external supports are not necessary, although there will be supports in the location between tokamaks to reduce the hoop force and support against gravity. The configurations may look as in Fig. 2. The hoop force could, of course, be eliminated almost entirely by the imposition of the vertical field. However, to depend on this even in the failure mode, requires an elaborate control system. Engineering trade-off study should be made to estimate this. In what follows, the physical parameter of a single coil whose major diameter is 150^{m} and 200^{m} and minor diameter of 5^{m} at $B_{max} = 9T$ is shown.

Table 2. Design parameter for the single coil.

	$2R_c = 150^m$	$2R_c = 200^m$
Current in coil (MA)	112.5	112.5
Hoop force (N/m)* = f_H	57.4×10^6	46.7×10^6
Cross sectional force = $f_H\ R_c$ (N)+	4305×10^6	4670×10^6
$f_H\ R_c/a_c^2$ (N/m^2)+	219×10^6	238×10^6
B_v (to make force free) (T)	0.51	0.42

* $\ln \frac{8R_c}{a_c} - \frac{3}{4} \sim \ln \frac{R_c}{a_c}$

+If not supported and no vertical field.

The single coil may be divided into 10 to 15 sections with a 20 m length/section used for the torus experiment and the remainder will be used as the support stations. Since at the support station, the minor diameter of the coil has no restrictive limit, structural supports and additional insulation could be provided. This, in turn, reduces the need for structural support where toruses are located. This should reduce the thickness Δ.

(2a) Advantages from the engineering point of view

The advantages from the engineering point of view are several. The accessibility is very much improved. Since there is no space limitation, the blanket could be placed at optimum distance from the plasma. Divertors both poloidal and bundle could be placed with minimum anguish. The maintenance should be easy, if the existence of the toroidal magnetic field will not interfere with the dismantling. (There are several ways to reduce the toroidal field, but none of them is completely satisfactory). We shall discuss these later. The design of the internal shield, i.e. between the toroidal field coil and the plasma is the exception for the above. We shall now discuss that.

(2b) The design of the internal shield.

We define internal shield as the shield between the plasma and the TF coil. This distance Δ is important in reducing B_{max} (hence the cost, presumably) as seen in Table 1.

There are several ways to reduce this distance Δ. First thing, one notes, is the solid angle sustained by the central column (Fig. 3). The fraction of the solid angle seen by the central plasma is $\frac{1}{\pi} \sin^{-1} \frac{R-a}{R}$. For the figure this fraction is approximately 23%. Now if we design an inner shield so that the shield is designed for heat removal and possibly neutron multiplications/reflection by employing Be, the shield design is greatly simplified. The poloidal coils are hopefully not needed in this area especially if the triangular shape is easier to make than the elliptical shape. (cf. the design of T.N.T.[5]). The neutran loss may be reduced to less than 20% by the use of Be and the natural shift of the plasma center towards the larger major radius.

As mentioned in Section 1, the plasma current may be established either by radial compression or translation parallel to the major axis.

(2c) Design of the single coil.

A large-diameter, superconducting coil is probably very difficult to construct. The design parameter of Table 1 is chosen to make the NbTi coil compatible with OCLATOR. The shrinkage on cooling will be taken up by the play left in the support stations so that the tension is minimized on cooling down. The dewar wall in the torus area may not be difficult to design as the support structures for the coil are not needed there.

In spite of these structural supports, probably it is prudent to make the coil (almost) force-free as it also makes the field more uniform. The second coil to take up the force may be placed at larger radius than R_c. Then this force-transfer coil (Fig. 2) has the current in opposite directions: hence if the force-transfer coil current decreases by some failure, the current in the single coil also decreases. The minor diameter of the force-transfer coil could be bigger: thus the material needs and maximum magnetic field are much lower.

(2d) Remote handling and maintenance.

Because there is a perfect access and because the TF single coil is most likely placed flat, 2 or 4 segments are indicated. Poloidal coils are to be segmented into 2 or 4. The removal, entirely remote, may not be too difficult if the torus is divided into two segments and pulled out on the rail not unlike the arrangement used in shipbuilding. It is probable that the vacuum building concept introduced by the ORNL TNS design could be used. Once the two or four segments are moved away from the radiation/strong magnetic field environment, further dismantling could proceed. The maintenance schedule and the system consideration will be further discussed in Section 4.

(2e) Some other possibilities.

(i) D-D reactor

If the single coil is installed, it may not be too difficult to change the size. In particular, if the ripple problem is less severe than currently believed, we may increase the size of the plasma. If we let $a = 5^m$, $R = 10^m$, $\Delta = 1.5^m$, $q = 1.2$, $B_{max} = 10T$ (at $a_c = 3.5^m$), $n\tau$ value could be ~20 higher than INTOR/ETF parameter. Thus D,D or D,D spiked with small amounts of T reactor may be practical.

(ii) Current-conducting inner-shield

If the inner shield is not going to breed tritium, it may be used to carry current to increment the toroidal field.

The amount of current to be carried is not large: otherwise the ohmic loss is intolerable. It appears, though, the current density of 0.5 ka/cm^2 may be permissible. If the area is roughly 5 m^2, the total current is 25MA. This increases the toroidal field by 20%. Alternatively B_{max} at the superconducting surface may be reduced by a factor 1.25. The ohmic loss with $\eta=4 \times 10^{-6}$ ohm-cm for the (major) axial length of 10^m would be 50 MW_e. Obviously this option should be evaluated with the cost trade-off.

(iii) Blanket, first wall and shield.

Since there is no space limitation and since frequent changeover is indicated, the blanket and the first-wall design should be relatively easy. Also the outer blanket need not have too high a shielding efficiency as there is no TF return legs to worry about. (Of course poloidal coils must be well protected.) The inner shield will be made by four layers: first wall, beryllium, shielding material, and copper or other conducting material. The latter two layers, as an option, will carry the axial current to boost toroidal field. Cooling pipes will carry the heat away.

Instead of designing large neutral loading in blanket, with the idea of changing the blanket often, and of possibly moving plasmas axially (Fig. 4), we can cut down the total neutron loading to less than 2MW-Y/cm^2. If so, ordinary material will probably suffice.

(2e) Magnetic shielding.

Obviously the major disadvantage is the existence of a magnetic field everywhere. The disadvantage is perhaps mitigated somewhat by the availability of space. The location of the neutral beam injector (if needed) could be farther than the conventional tokamak system. Even so the shield against at least 1 T field is needed.

(2f) The energy storage.

The total energy of the single coil, if the force transfer coil (Fig. 2) is not energized, is approximately 3000 GJ or approximately 1 million kWh. While this energy is large it is not impractical. As a matter of fact, this coil could be used as the energy storage facility for the 1 GW generating plant.

3. Other physics considerations.

The configuration of OCLATOR is suitable for compression and axial translation. The advantage of that configuration has been described before[6].

Poloidal divertors or even bundle divertors could be placed in this configuration. If the axial placing of main reacting plasmas is to be moved as in Fig. 4, locating a single divertor flexible enough to accommodate all different positions might be impossible. By placing two or three divertors, however, several plasma positions may become compatible with the divertor.

The ripple consideration may be greatly altered if a single coil produces the field ripple. Especially if q is less than 2, the trapping in the ripple is actually comparable to the (minor) axial length of the banana orbit. Hence higher ripple may well be allowed. Some theoretical estimate is called for.

D-shape plasma is more resistant to the positional instability than elliptical plasma. Thus $g(\delta)$ could be raised to 1.5 without resorting to the complicated feedback stabilization. Again this calls for the theoretical studies.

4. System consideration

The design of OCLATOR depends on the construction of many toruses: hence the startup cost is high. On the other hand, the trend of siting of power-plants tends to be fewer sites with higher power output/site for nuclear power plants. Hence in the future to have 7 tokamaks generating 1GW/tokamak may not be too outrageous. Especially if there is an international agreement and the output is used to produce chemical material (e.g. fertilizer) or synthetic fuel, the objection to large power concentration may not be too serious.

The concentration of power plants has its advantages. For example, inventory of spare parts could be reduced, more trained personnel and backups are available. The power transmission line could be constructed with the maximum efficiency with minimum environmental objections etc. Most importantly, we can increase the frequency of maintenance.

If there are ten tokamaks attached to the single TF coil system, (OCLATOR), then 7 of them could be operational at a given time with one used as a spare in case of failure, while one is readied for the start-up and another being dismantled.

The thermal power will be fed to several generating stations with a couple of million kw fossil-fuel power plants as the backup. The thermal pollution and environmental consideration may suggest that this kind of power island could be located in a cold climate. (However the power generated in Manhattan Island alone could easily compete with 20 ~ 30 GW_{th} power plant considered here).

Acknowledgements

Discussions with R. Derby (ORNL ETF), P. Bonanos, and D. Jassby are greatly appreciated. Especially R. Derby for emphasizing the idea of a single coil with many tokamaks attached. D. Jassby gave the author his vast knowledge on reactor designs, including SMARTER, hybrid, and LPSX.
This work was supported by Department of Energy contract number EY-76-C-02-3073.

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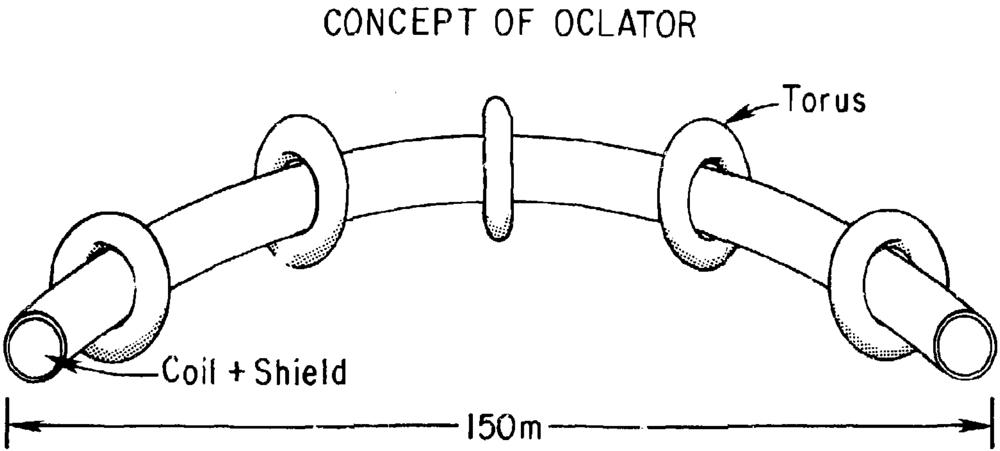

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Fig. 1. Concept of OCLATOR.

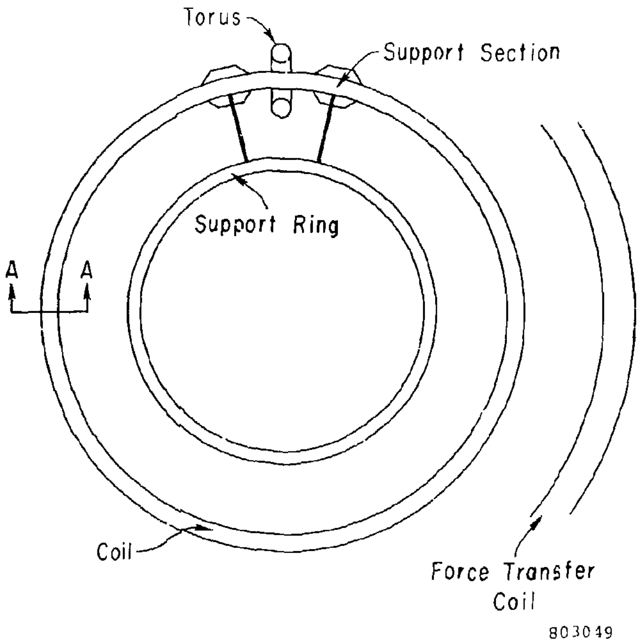

Fig. 2. Plane view of the single toroidal field coil.
(not drawn to scale.)

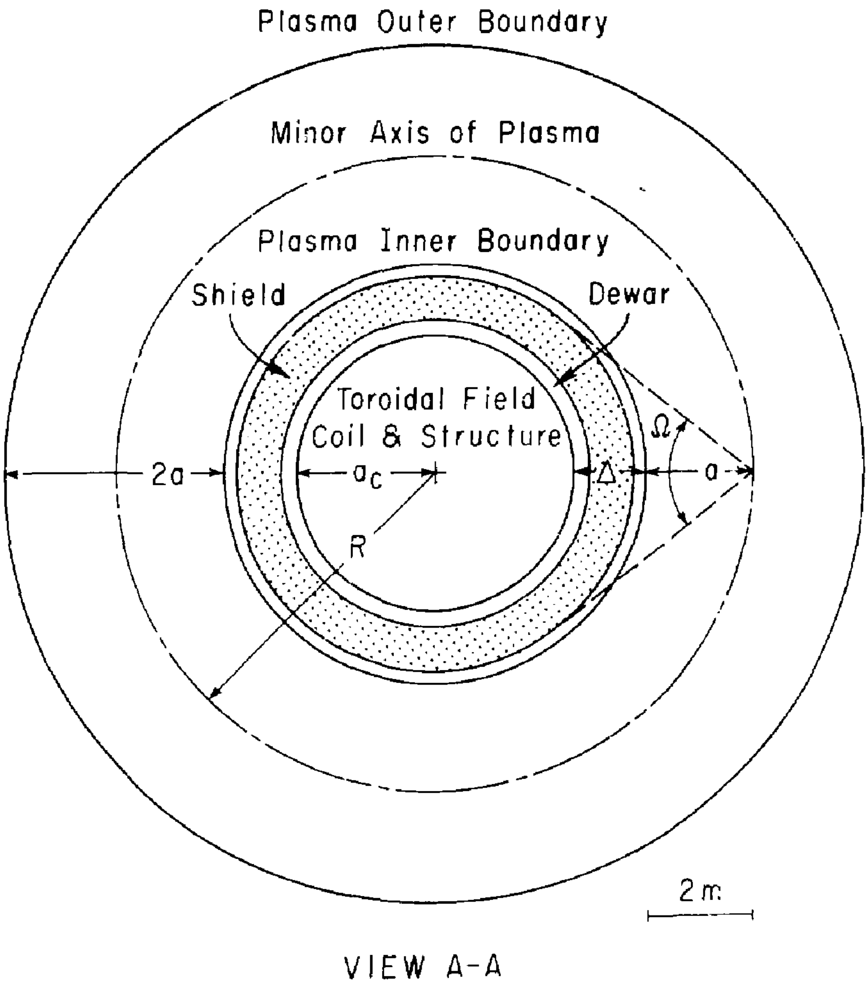

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Fig. 3. Plane view of torus. The view is perpendicular to Fig. 2.

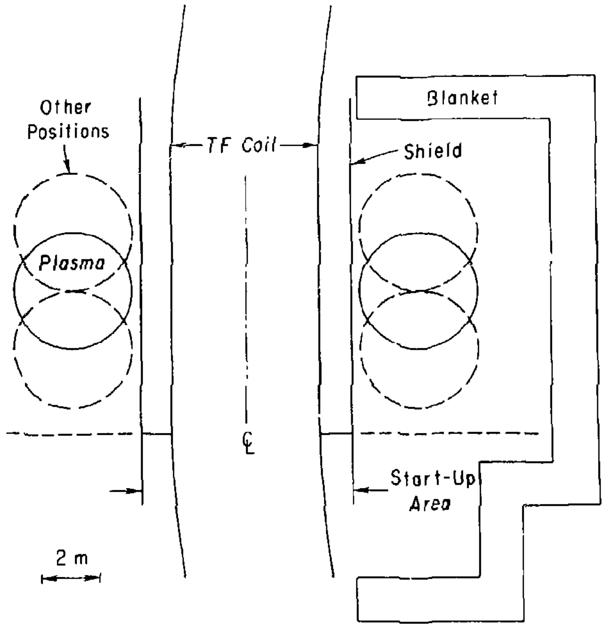

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Fig. 4. Cross sectional view. Orientation same as Fig. 2 and perpendicular to Fig. 3.