J Fusion Energ
DOI 10.1007/s10894-010-9303-1

ORIGINAL RESEARCH

Why Compact Tori for Fusion?

S. Woodruff · M. Brown · E. B. Hooper · R. Milroy · M. Schaffer

Abstract A compact torus (CT) has a toroidal magnetic and plasma geometry, but is contained within a simply-connected vacuum vessel such as a cylinder. Spheromaks and field-reversed configurations fall into this category. Compact tori are translatable and have a high engineering beta. The primary benefit of CTs for fusion is the absence of toroidal field and Ohmic Heating coils and the many problems brought on by them. Studying fusion-relevant plasma in simply-connected geometries affords the world fusion program both physics and technology opportunities not found in other configurations. This paper outlines the technology and physics opportunities of compact tori, and presents a cost model based on geometry for comparison with less compact configurations.

Keywords Compact tori · Spheromak · Field reversed configuration

S. Woodruff (✉)
Woodruff Scientific Inc, 4501 Shilshole Ave NW, Seattle, WA 98107, USA
e-mail: simon@woodruffscientific.com

M. Brown
Swarthmore College, 500 College Ave, Swarthmore, PA 19081, USA

E. B. Hooper
Lawrence Livermore National Laboratory, 7000 East Ave, Livermore, CA 94550, USA

R. Milroy
Redmond Plasma Physics Lab, University of Washington, 14700 NE 95th Street, Redmond, WA 98052, USA

M. Schaffer
General Atomics, PO Box 85608, San Diego, CA 92186-5608, USA

Introduction

In the next 5–15 years, two devices (NIF and ITER) will produce ignited plasmas, the next steps from which will be demo (power-producing) reactors. As we enter the era of NIF and ITER, several concepts are being developed in parallel that offer opportunities for resolving well known critical issues. Within magnetic fusion, the concepts known as 'Compact Tori' are researched: plasma toroids that have no material linking the plasma. Removing the need for toroidal field (TF) coils means that the resulting configuration can be compact and highly modular, lowering cost and providing easier maintenance. Without an externally imposed toroidal field, compact torus (CT) plasmas are stabilized either by appropriately tailoring the profile of currents flowing in the plasma or by the presence of a population of highly kinetic ions, allowing operation at high beta. Formation and current drive are achieved by a variety of novel techniques involving magnetic reconnection that now are finding application for non-inductive start-up in larger machines. CTs therefore offer many unique opportunities for resolving critical issues relating to both technology and plasma physics, and serve as valuable test-beds for the development of new ideas.

The ideas presented here form a distillation of thoughts relating to CTs from two recent DOE planning activities: Fusion Energy Sciences Advisory Committee (FESAC) Toroidal Alternates Panel (TAP) [1] and The Burning Plasma Organization Research Needs Workshops (ReNeW) [2]. The FESAC TAP report defines the Compact Torus concepts in great detail, and states the ITER era goal: "To demonstrate that a CT with simply connected vessel can achieve stable, sustained or long pulsed plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning CT plasma experiment." In the report,

Published online: 12 June 2010

three primary challenges are outlined for the ITER era: (1) formation/stability in the reactor-relevant regime, (2) anomalous transport/energy confinement, (3) efficient current drive/flux sustainment. To remain a viable alternative, these three challenges will need to be addressed in next step CT experiments. The follow-on ReNeW report sketches the critical development needs for each concept, with great synergy with other concepts such as the ST and RFP. An expanded Compact Torus section of the *ReNeW Report* will be published in part as a paper [3], and will discuss the prioritization of research activities by outlining technical road-maps. We refer the reader with interest in understanding the critical physics issues to these reports.

This paper is structured as follows. The next section "Technology Opportunities" discusses simplified geometry, reduced cost, and increased reliability/availability and easier maintenance. The section on "Physics Opportunities" outlines the physics opportunities for the class of concepts known as CTs (Spheromaks and FRCs). Discussion and Summary follow.

Technology Opportunities

Simplified Geometry

Figure 1 illustrates the basic premise of CTs. While still needing to burn deuterium and tritium, and hence needing a blanket to breed tritium and capture energy released in the form of neutrons, the TF and Ohmic Heating (OH) coils are absent, thereby reducing complexity of a fusion reactor. The simply connected boundary and absence of TF coils provide a natural divertor with unobstructed plasma exhaust to external divertor targets. In this way, wall and divertor loadings can be significantly decoupled in CTs: divertors can be protected from neutron bombardment and exhaust heat does not have to be absorbed by the plasma chamber. The simply-connected nature of CTs also potentially allows easier implementation of such advanced technologies as liquid metal walls and remote maintenance of components. The fact that a CT can be translated offers flexibility to reactor design. For example, a CT can be formed in one chamber, and then translated to a second confinement/burn chamber, thus reducing the radiation exposure to the formation region.

Reduced Cost

To illustrate the cost savings for a compact geometry, a model based on the geometries shown in Fig. 2 was developed, shown in full in the Appendix.[1] The model focuses on the importance of the simple geometry of the CTs to achieve significant savings in complexity, reliability

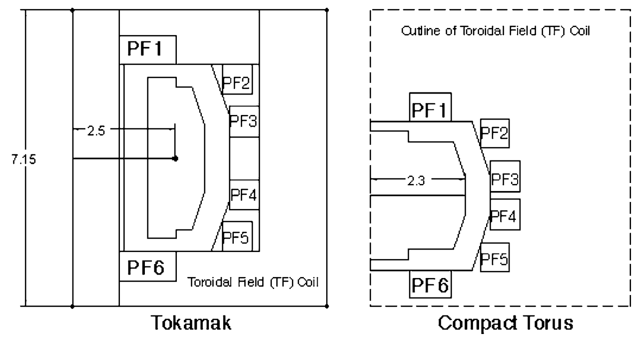

Fig. 1 A tokamak is shown next to a configuration with the same surface area without the TF and OH coils. The resulting configuration is called a 'Compact Torus.' Dimensions shown have units of meters

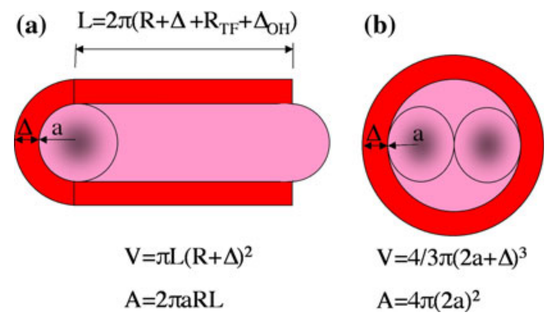

Fig. 2 Simplified geometries for **a** the tokamak and **b** spheromak blanket and shields. In the case of the tokamak, a tube is the most appropriate geometry, and for the CT, a sphere

and costs. It is assumed that the added costs of auxiliary power and current drive systems are similar for the tokamak and CTs, as is the costs of the conventional power producing equipment such as the steam or metal heat-carrying fluids and turbines. In the case of a tokamak, a large-aspect-ratio geometry is chosen, modeling the blanket/shield as a tube, Fig. 2a), and for the CT, a spherical geometry is used [4, 5]. It is taken as axiomatic that (1) fusion-grade steady-state plasmas are achievable in CTs; (2) the fusion power from either system is the same, given by systems of identical surface area, and; (3) a maximum wall loading (neutrons, particles and radiation) of 5 MW/m^2 is achieved. As a comparison, the physical dimensions of the Fusion Development Facility (FDF) [6] are used ($a = 0.7$ m, $R_{\rm TF} = 1.08$ m, $d_{\rm sol} = 0.2$ m) and a spherical geometry with the same surface area is used in the case of the CT.

The consequences of geometric simplicity are significant. Figure 3 shows the dependencies of (a) cost and (b) mass power density (MPD)[2] on blanket thickness for two systems. The trend of cost with blanket thickness is almost

[1] The TF, OH and PF coils are also modeled, but the divertor is omitted to simplify the discussion.

[2] MPD is an important measure of economy for a fusion power core, see [7] for a discussion.

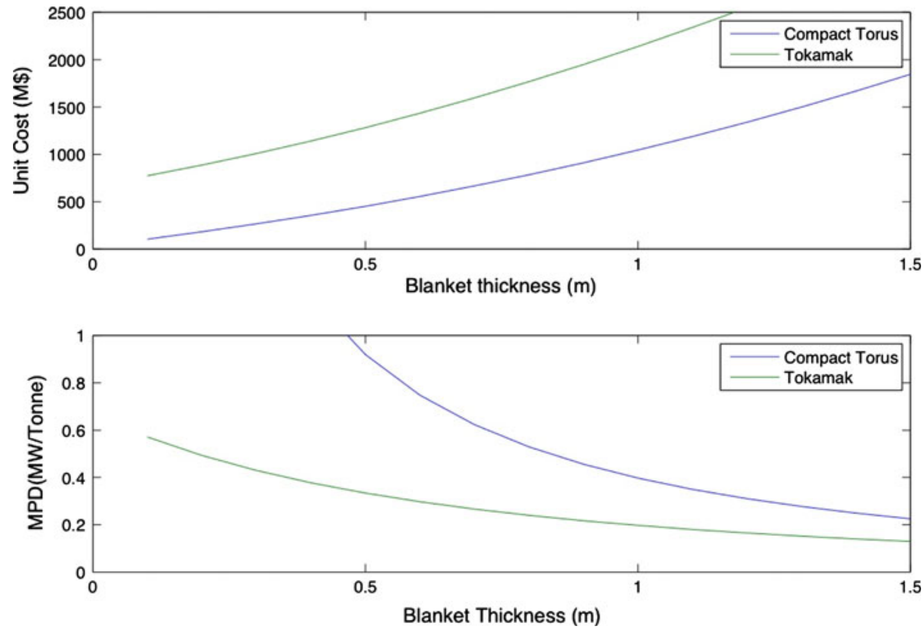

Fig. 3 Comparison of dependencies for a 300 MW CT and tokamak for **a** power per unit cost with blanket thickness; **b** mass power density with blanket thickness

identical for the CT and tokamak, although is offset in the case of the tokamak by the cost of the additional coils, as one might expect. Most commercial reactor studies conclude that the blanket and shield must be >1 m thick for compatibility with superconducting coils, without which the power consumed by copper coils would be too great. In the plot of the MPD for the two systems it is shown that even with a blanket thickness greater than 1 m, there is a perceptible cost saving (by a factor of 2) by omitting TF and OH coils.

Increased Reliability/Availability and Easier Maintenance

By omitting TF coils, the blanket, shield and coils can be much simpler, particularly for demounting. Figure 4 shows the blanket, coil and shield (4a) and an expanded view of annular blanket module sections (4b). Without the need to demount chamber-linking (or plasma-linking) coils, the system becomes much easier to disassemble and time for repairs becomes much shorter ultimately allowing for greater availability. Blanket fluid flow is simple: metallic fluid flow will be parallel to the equilibrium magnetic field. Flow patterns for liquid first walls would also be parallel to applied field.

Physics Opportunities

CT Variants

The class of concepts known as Compact Tori entail both the spheromak and the Field Reversed Configuration (FRC). However, the spheromak is related most closely to the Reversed Field Pinch by virtue of a q-profile that has reversed magnetic shear everywhere and falls within the range from 0.2 to 1; and also by forming plasmas that are close to the Taylor state. The FRC in contrast has no (or very weak) toroidal magnetic field, and is stabilized instead by highly kinetic ions. The concepts are sketched in Fig. 5a); and, example equatorial magnetic field profiles are provided in 5b). Profiles are derived from analytic equilibrium models for the spheromak [8] and for the FRC [9]. Understanding and demonstrating the physics required to make the CTs successful fusion concepts needs considerable more resources than have been allocated to date. In particular, stable operation at a safety factor less than unity needs further development. However, if it can be achieved, the payoff for fusion energy development is significant as described in this report. The focus of Refs. [1, 2] is on a research plan to achieve this goal.

Present day CTs are at the 0.1 m minor radius scale with magnetic fields up to 1 T. CTs tend to operate at high beta (=plasma pressure/magnetic pressure), so the physics of high-beta fusion-relevant plasmas is readily studied in CTs. The strongly self-organized nature of CTs presents unique plasma physics. Some present day CTs operate in an interesting kinetic regime wherein the ion Larmor orbit is a substantial fraction of the machine size. Kinetic effects are believed to play a key role in stabilizing these plasmas, however for most CT reactor concepts a smaller ion orbit is envisioned. Finally, formation of CTs often involves complex, dynamical relaxation processes of general interest to plasma physics and fusion.

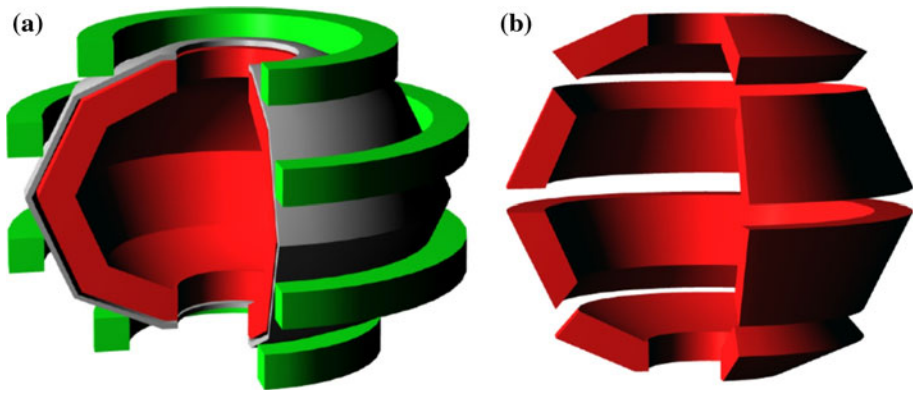

Fig. 4 Blanket and shield construction. **a** Non-interlocking coils and simply-connected blanket/shield of the CT fusion power core; **b** It is anticipated that blankets can be modular and built from annular sections, providing ease of maintenance and increasing availability by reducing down-time

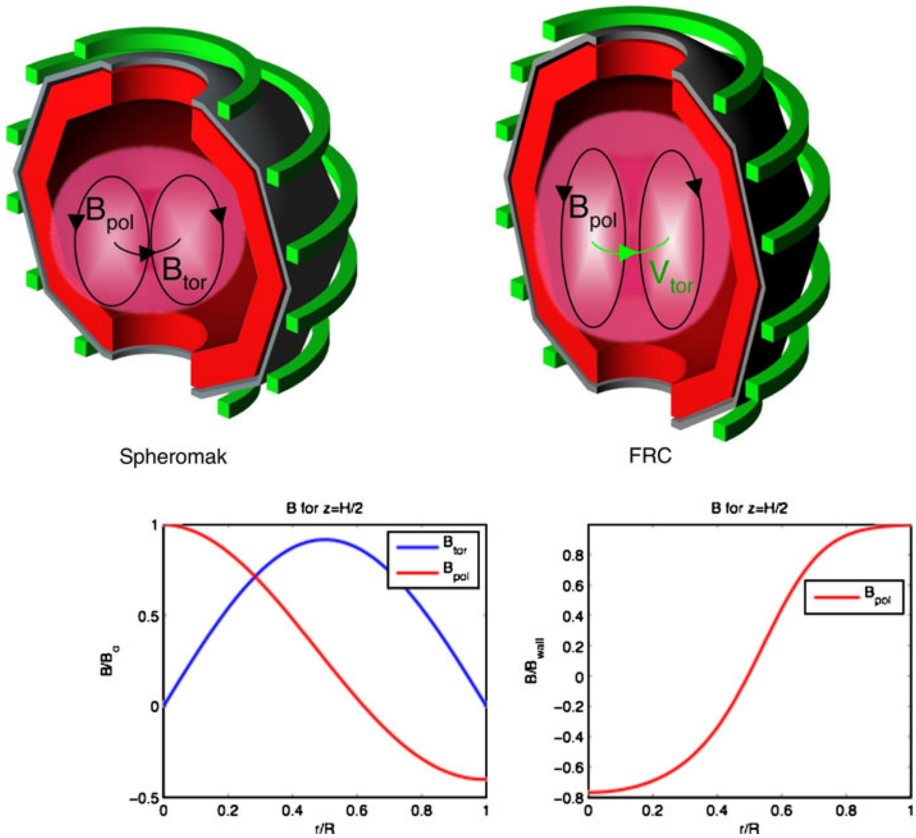

Fig. 5 Sketches of the Spheromak and Field Reversed Configuration. **a** Magnetic topology depends strongly in each case by the shape of the flux-conserving boundary; **b** Internal magnetic field profiles in the midplane

Spheromaks

By careful attention to vacuum quality, and by rudimentary current profile control to suppress turbulence, spheromaks obtain peak $T_e = 0.5$ keV and core energy confinement similar to L-mode in tokamaks. Peak electron betas above 20% have been obtained. By methods of pulsed and continuous helicity injection, non-inductive startup and sustainment of mega-ampere plasma currents have been demonstrated. Resistive MHD simulations, developed for the tokamak, have been used to understand spheromak physics, interpret experimental results, and now are used to help design new devices. Spheromaks have also been used in studies of basic plasma physics, including magnetic reconnection and the generation of energetic particles during reconnection.

FRCs

FRCs formed in theta-pinches obtained $\sim 10^{21}$ m^{-3} densities, keV ion temperatures, and high beta ($\beta > 0.5$). Currents are mainly diamagnetic (resulting from dp/dr), although weak toroidal fields are sometimes observed. Many experiments made pulsed, prolate FRCs by formation in theta-pinches. Oblate and weakly prolate (elongation $E < 2$) FRCs have been formed by merging spheromaks where oppositely directed toroidal magnetic fields annihilate, transferring their magnetic energy into electron and ion kinetic energy. Rotating magnetic field (RMF) current drive has formed and sustained prolate FRCs.

Discussion

To capture the opportunities presented by simpler geometry, much work still remains for CT concepts to attain the ITER era goal of demonstrating "… that a CT with simply connected vessel can achieve stable, sustained or long pulsed plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning CT plasma experiment." The required research, however is mostly agreed upon, and pathways directed to overcoming technical hurdles are mapped out (see accompanying paper).

We have so far dwelled on a CT concept that, while dissimilar in terms of the coils, is similar in every respect to the reigning tokamak concepts, namely: all anciliary subsystems are assumed to be the same. This need not be true: the CT concepts each have unique and often simpler non-inductive current drive schemes (that are now finding employment in larger tokamaks); in the case of the FRC, plasma heating is often caused by adiabatic compression during formation; in the case of the divertors there is a great dissimilarity, wherein the CT divertors are not constrained spatially by the TF coil and incident power can be such that today's materials suffice; finally, the energy conversion systems may ultimately differ greatly, whereby direct energy convertors (with greater efficiency than thermal cycles) are appropriate and feasible due to geometry.

The geometrical cost model points to some perhaps obvious areas for improvement. To start, it is assumed that surface power loading of 5 MW/m^2 cannot be exceeded. This incident power, however, is not separated into plasma heat, radiation and neutron fluxes. In the CT geometry, it is possible to send heat flux to a divertor mounted outside the main chamber, thereby increasing the total allowable neutron and radiation fluxes on the first wall. Given similar neutron and heat fluxes, it may be possible to increase the fusion power (P_f) for a given system size. Other factors could be gained by more accurate modeling of the component cost and possibly use advanced fuels, which could obviate the blanket, but places increased demand on confinement (see [10] for a discussion and follow references to Rider in particular). The same model is also useful for considering pulsed systems: for a system of the same size and average power as steady-state reactor, pulsed systems will require instantaneously higher wall loadings. The means for handling such high power loadings remains an open area of materials research. Finally, if practicable, smaller scale reactors (<GWe) based on the CT concept could well be a more attractive end product for private sector development, though such concepts presently lack a physics basis.

Given similar performance to tokamaks, the model presented here shows that, by omission of TF and OH coils, the cost of a burning plasma device can be reduced by a significant factor. However, before even reaching burning plasma conditions, the omission of coils will reduce the unit cost (relative to tokamaks) of next step CT devices at the Proof of Principle and Performance Extension stages. CT researchers also draw benefit from an enormous transfer of knowledge from tokamak science and technology, and in particular from the ability to simulate dominant plasma phenomena, thereby allowing larger more confident strides to be taken in parameter space, and thereby reducing development costs further. The present-day CT therefore offers a tantalizing development path for exploring high performance plasmas at significantly lower cost.

Of course, the opportunities outlined here can only be captured if the scientific issues of sustainment (steady-state is assumed) and confinement (tokamak-like confinement is assumed) can be addressed in a timely manner. Given the recent (decadal) progress in CT research and performance, we strongly believe that the most recent planning activities contain all of the necessary steps to meet all of the scientific requirements. CTs therefore represent the possibility of fundamentally changing the game.

Summary

The Compact Torus represents a radical design change for magnetic fusion systems: one in which the cost of a burning plasma experiment could be reduced significantly by omitting Toroidal Field and Ohmic Heating coils, giving a lower cost and more compact fusion power core that is both easier to maintain and hence provides greater availability and reliability. After two particularly intensive planning activities (FESAC TAP and BPO Renew), the critical issues are well defined for CTs and clear roadmaps for addressing these issues are available. It is expected that in the next 20 years CTs will achieve stable, sustained or long pulsed plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning plasma experiment.

Acknowledgments We acknowledge John Sheffield, and Farokh Najmabadi for outlining the starting point for the discussion of cost and to Ron Miller for indicating where it might go. Thanks also to Charlie Baker for encouraging this line of thinking. Work by E. B. Hooper was performed under the auspices of the US Department of

Energy by Lawrence Livermore National Laboratory under contract DE-AC52-07NA27344. S. Woodruff performed this work while supported by Department of Energy under subcontract numbers DE-FG02-06ER84449 and DE-FG02-07ER84924.

Appendix: Matlab Cost Model

```
% economics.m:
%
% last modified:                        2009 July 18th by S. Woodruff
% =====================================================================
% WRITTEN BY:  Simon Woodruff       on       July 18th 2009
%
%                     FREE FOR DISTRIUBUTION
% =====================================================================
%

%GLOBAL PARAMETERS----------------------------------------------------------

P_w = 5;                    %Maximum wall loading [MW/m^2]
k=1;                        %Cost factor [M$/tonne]
m=10;                       %Mass factor (Tonne/m^3)
eff=0.4;                    %Efficiency of plant

%GEOMETRY-------------------------------------------------------------------

%tokamak
R=0.7;                      %Minor radius of first wall in FDF [m]
r_tf=1.08;                  %Radius of inner TF leg in FDF [m]
d_sol=0.2;                  %Radial thickness of solenoid in FDF [m]
N=2*pi;                     %Multiplier for length
L=N*(R+d+r_tf+d_sol);       %Length of tube [m]
d=0.5;                      %Thickness of blanket [m]
L_pf=0.4;                   %Length of one side of PF coil [m]

%compact torus
R_ct=2.3;                  %Major radius of 1st wall [m]

%GEOMETRICAL RELATIONS------------------------------------------------------

    %SURFACE AREA OF 1st WALL
    A_s      = 2*pi*R*L;                    %[m^2]
    A_s_ct   = 4*pi*R_ct^2;                 %[m^2]

    %VOLUME OF PLASMA
    V_p      = pi*R^2*L;                    %[m^3]
    V_p_ct   = 4/3*pi*R_ct^3;               %[m^3]

    %VOLUME OF BLANKET
    V_b      = pi*(2*R*d + d^2)*L;          %[m^3]
    V_b_ct   = 4/3*pi*((R_ct+d)^3-R_ct^3);  %[m^3]

    %VOLUME OF BLANKET AND PLASMA
    V        = pi*L*(R+d)^2;                %[m^3]
    V_ct     = 4/3*pi*(R_ct+d)^3;           %[m^3]

    %VOLUME OF PF COILS
    V_pf     = 6*L_pf^2*2*pi*(2*R+2*d+r_tf+d_sol);            %[m^3]
    V_pf_ct = 6*(L_pf/2)^2*2*pi*(R_ct+d);        %[m^3]

    %VOLUME OF TF COIL
    V_tf     = 4*pi*r_tf^2*(2*R+2*d+2*r_tf);  %[m^3]

    %MASS OF MACHINE
    M        = m*(V_b+V_pf+V_tf);            %[Tonne]
    M_ct     = m*(V_b_ct+V_pf_ct);           %[Tonne]

%CALCULATE PARAMETERS, P_f, P_den, P_mpd for a point design---------------

    %FUSION POWER
    P_f      = A_s*P_w;                    %[MW]
    P_f_ct   = A_s_ct*P_w;                 %[MW]

    %POWER DENSITY OF BLANKET + PLASMA
    P_den    = P_f/V;                       %[MW/m^-3]
    P_den_ct= P_f_ct/V_ct;                  %[MW/m^-3]

    %MASS POWER DENSITY (MPD)
    P_mpd    = P_f/M;                       %[MW/Tonne]
    P_mpd_ct= P_f_ct/M_ct;                  %[MW/Tonne]
```

```
    %CAPITAL COST OF POWER CORE
    C        = k*M;                             %[$M]
    C_ct     = k*M_ct;                          %[$M]

    %POWER PER UNIT COST
    PC       = P_f/C;                           %[MW/$]
    PC_ct    = P_f_ct/C_ct;                     %[MW/$]

    %POWER OUTPUT
    P_out    = eff*P_f;                         %[MWe]
    P_out_ct= eff*P_f_ct;                       %[MWe]

%WRITE OUT ALL OF THE PARAMETERS-------------------------------------------

    Out1 = {'Plasma Volume' 'Machine Volume' 'Fusion Power' 'Power Density' 'MPD' 'Cost'
'Power/Cost' 'P_out'}
    Out2 =  [ V_p V P_f P_den P_mpd C PC P_out; V_p_ct V_ct P_f_ct P_den_ct P_mpd_ct C_ct
PC_ct P_out_ct]

%GENERATE DEPENDENT FUNCTIONS--------------------------------------------

    for i=1:10000,
    R2(i)=i/10;
    D(i)=i/10;
    L2(i)=N*(R2(i)+r_tf+d_sol);

    %FUSION POWER
    P_f(i)       = 2*pi*R2(i)*L2(i)*1.25*P_w; %MW/m
    P_f_ct(i)    = 4*pi*R2(i)^2*1.25*P_w; %MW/m

    %POWER DENSITY OF BLANKET + PLASMA
    P_den(i)     = (2*pi*R2(i)*L2(i)*1.25*P_w)/(pi*L2(i)*(R2(i)+D(i))^2);  %MW/m^-3
    P_den_ct(i) = (4*pi*R2(i)^2*1.25*P_w)/(4/3*pi*(R2(i)+D(i))^3);  %MW/m^-3

    %MASS POWER DENSITY (MPD)
    P_mpd(i)     = (2*pi*R*L*1.25*P_w)/(m*(pi*L*(2*R*D(i) +
D(i)^2)+6*L_pf^2*2*pi*(R+D(i)+r_tf+d_sol)+4*pi*r_tf^2*(2*R+2*D(i)+2*r_tf))); %MW/Tonne
    P_mpd_ct(i) = (4*pi*R_ct^2*1.25*P_w)/(m*(4/3*pi*((R_ct+D(i))^3-
R_ct^3)+6*(L_pf/2)^2*2*pi*(R_ct+D(i)))); %MW/Tonne

    %COST
    C(i)         = k*(m*(pi*L*(2*R*D(i) +
D(i)^2)+6*L_pf^2*2*pi*L/N+4*pi*r_tf^2*(2*R+2*D(i)+2*r_tf)));
    C_ct(i)      = k*(m*(4/3*pi*((R_ct+D(i))^3-R_ct^3)+ 6*(L_pf/2)^2*2*pi*(R_ct+D(i))));
    end

%PLOTTING---------------------------------------------------------------
    figure(1)
    %PLOT COST AS A FUNCTION OF BLANKET THICKNESS
    subplot(2,1,1)
    plot(D,C_ct, D,C)
    axis([0 1.5 0 2500])
    xlabel('Blanket thickness (m)')
    ylabel('Unit Cost (M$)')
    legend('Compact Torus', 'Tokamak')
    %PLOT MASS POWER DENSITY AS A FUNCTION OF BLANKET THICKNESS
    subplot(2,1,2)
    plot(D,P_mpd_ct,D,P_mpd)
    axis([0 1.5 0 1])
    xlabel('Blanket Thickness (m)')
    ylabel('MPD(MW/Tonne)')
    legend('Compact Torus', 'Tokamak')
```

References

1. FESAC TAP report David Hill (Chair), Rich Hazeltine (Vice Chair) http://fusion.gat.com/tap/final_report.php/
2. BPO ReNeW report Rich Hazeltine (Chair). David Hill (Vice Chair) http://burningplasma.org/web/renew.html
3. E.B. Hooper et al., 'The Compact Torus: Spheromaks and FRCs' white paper in preparation for this Journal
4. J. Sheffield, Private communication with corresponding author, June 2009
5. F. Najmabadi, Private communication with corresponding author August 2009
6. R. Stambaugh, 'Overview of the Fusion Development Facility' presentation given at the 2–4 March 'Harnessing Fusion Power' workshop, UCLA available at http://www.fusion.ucla.edu/FNST/Renew_Presentations/
7. R.A. Krakowski, 'Simplified Fusion Power Plant Costing: A General Prognosis and Call for "New Think"' Fusion Technology, March 1995, Volume 27
8. P.M. Bellan, Phys. Plasmas **9**, 3050 (2002)
9. L.C. Steinhauer, Phys. Fluids **B2**, 3081 (1990)
10. W.M. Nevins, A review of confinement requirements for advanced fuels. J. Fusion Energ. **17**(1), (1998)