A BUNDLE DIVERTOR FOR INTOR

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1. INTRODUCTION

Studies undertaken for INTOR indicate that for a remotely handled, ignited tokamak both "conventional" poloidal divertors as (PDX, Phase 1) and bundle divertors (ISX-B, Phase 1) will require improvement. Work has been underway for more than a year at ORNL in collaboration with Westinghouse on improving the bundle divertor concept. This report summarizes the work to date.

Divertor Operation

A divertor may perform a number of functions in the tokamak system.

- It serves as a magnetic limiter which constrains the size of the main body of plasma without allowing it to come into contact with the walls of the discharge chamber.

- It provides a shielding layer of plasma which ionizes a large fraction of the impurities generated at the wall and sweeps them out of the chamber.

- It removes some of the particles which cross the separatrix and limits the hydrogen recycling at the wall. This means that it can remove helium generated in the body of the plasma provided it diffuses out. In addition hydrogen may be added by gas and pellet injection to maintain the particle balance and this gives some external control of the edge region.

- It removes heat reaching the plasma edge to targets outside the main plasma chamber where in principle it may be handled more readily.
The purpose in writing these functions down again is to place some priority on them for INTOR

- For long pulses > 20s helium must be removed.
- We must prevent the build-up in the plasma of wall generated impurities.
- To prevent wall generated impurity build-up we probably need control of the edge region.
- This at a minimum would seem to require that we may add and subtract hydrogen in a controlled fashion.
- We need to control the heat coming from the plasma.

From this somewhat ad hoc prioritizing we see that the ability to remove heat is in some sense the least of our worries. In fact at typically 40 W cm$^{-2}$ thermal power crossing the separatrix it would not be difficult to take this load on the first wall of the torus. A problem in the divertor field is to establish a self-consistently calculated regime in which a satisfactory balance is achieved between the above functions.
2. THE BUNDLE DIVERTOR - PROBLEMS AND SOLUTIONS

This divertor has the advantage that in principle it may be installed between two adjacent toroidal coils in a manner that would allow it to be replaced as a module. This is clearly of major importance for application to a remotely handled, ignited tokamak. In fact to make such a plug-in divertor requires that the total tokamak system is designed to permit it. One of the critical issues for the INTOR device is to establish the form of the divertor so that the device may be designed.

A possibly bad feature of the divertor is that since it acts to form a separatrix in the toroidal field it leads inevitably to a large local toroidal field ripple in the body of the plasma. There are, in fact, a variety of "good" and "bad" features of this divertor and the goal of the research program is to study them and evolve a satisfactory optimized design which represents an INTOR relevant compromise.

- The aperture in the wall for the extracted flux bundle is generally a small fraction of the plasma surface area $\leq 10^{-2}$. In consequence the extracted power density may be high, i.e., for typical reactor level thermal power to the plasma edge $\sim 40 \text{ W cm}^{-2}$, the average power level in the diverted bundle may be $\gtrsim 4 \text{ kW cm}^{-2}$. This level of power is undesirably high for the divertor target. Various remedies have been proposed to deal with the problem including tilting the target and spreading out the flux bundle with careful design of the basic divertor coils Yang\textsuperscript{(1)} with additional exhaust coils Stott and Sanderson\textsuperscript{(2)}. One of the issues is how much of the power and particle flux leaving the plasma will reach the divertor. Approximately self-consistent calculations of this for an INTOR scale plasma are presented in Section 3.
Roughly one-third of the power and particle flux reaches the divertor for the cases considered, Houlberg and Attenberger\(^{(3)}\).

- The coil current densities are large and the stresses, both thermal and mechanical, are considerable. For example, in the DITE MK1 divertor the coil current density was \(\sim 20 \text{ kA cm}^{-2}\) for a toroidal field of 1.8T. In fact, as discussed by Crawley\(^{(4)}\), the divertor coil current density scales favorably with increasing machine size.

For constant toroidal field if the tokamak and divertor are scaled as

\[
\frac{R_{\text{new}}}{R_{\text{old}}} \quad \begin{align*}
\text{Coil power} & \quad \alpha k \\
\text{Current density} & \quad \alpha k^{-1} \\
\text{Power density} & \quad \alpha k^{-2} \\
\text{Forces} & \quad \alpha k^2 \\
\text{Stresses} & \quad \alpha k^0
\end{align*}
\]

The problem is to make a satisfactory trade-off between the power and stresses and the ripple in the plasma. For an INTOR like device the divertor coils must include space for neutron shielding. It has been suggested that about 40 cm would be required for water-cooled copper coils which would have to be replaced every 2 years. Designs of 2-coil and 4-coil divertors with shielding are discussed in Section 4, Yang\(^{(5)}\).

The results of a brief study of ways of installing a modular water-cooled copper divertor in a tokamak with superconducting coils are given in Section 6, Nelson and O'Toole\(^{(6)}\). The impact of a bundle divertor on the choice of the size and field of INTOR is discussed in Section 7.
• The bundle divertor leads to increased and localized toroidal field ripple at all radii. In the basic 2-coil design of DITE and ISX-B the ripple on axis is \( \frac{B_{\text{max}} - B_{\text{min}}}{B_{\text{max}} + B_{\text{min}}} \approx 2\% \). A coil arrangement which leads to \( \approx 0.5\% \) ripple is obtained with 4 symmetrically placed divertor coils. Preliminary studies of magnetic ripple effects on fast particle confinement are presented in Section 5, Rome and Fowler\(^{(7)}\).
References

1. T. F. Yang et al., 8th Symposium IEEE, 1979


3. PARTICLE AND HEAT BALANCE IN INTOR SCALE IGNITED PLASMAS

W. Houlberg and S. Attenberger (ORNL)
Summarized by J. Sheffield

Preliminary computations of an INTOR plasma from start-up and including the burn phase have been made using a full radial 1-D code which includes a bundle divertor and scrape-off layer. The machine parameters are:

\[ a = 120 \text{ cm}, \quad R = 480 \text{ cm}, \quad b/a = 1.6 \]
\[ B = 5.3 \text{ T}, \quad I = 4 \text{ MA} \]

The scrape-off layer is 10 cm wide, and the divertor extends over \(1/10\) of the azimuthal boundary, i.e., it is 80 cm high. The width of the bundle through the wall is 20 cm. During the burn phase the plasma is maintained by fusion \(\alpha\)-power. The fusion power remains approximately constant as a result of maintaining a constant average density of both D and T.

There is a steady gas puff rate of D, with no recycling, i.e., no distinction is made between puffing and recycling. There is feedback separately on the injection of D and T pellets to maintain constant \(\langle n \rangle\) in the central plasma out to the separatrix. The pellets are 2 mm radius at \(2 \times 10^5 \text{ cm}^2 \text{s}^{-1}\). The program runs for a simulated time of 20 to 30s. The transport coefficients are as given in INTOR Report 19, by Singer, Post and Rutherford with the addition of ripple losses. The ripple losses act to hold the temperature down.

In the present version there is not quite an exact particle balance because with the fluctuations induced by the pellet injection it is difficult to do an exact accounting. Nevertheless these computations indicate broadly the kind of particle flux and heat load to be expected.
in the divertor. The plasma parameters are:

\[ \langle n_e \rangle \approx 1.15 \times 10^{14} \text{ cm}^{-3} \]
\[ \langle T_e \rangle = 25 \text{ keV} \]
\[ \langle T_i \rangle = 24 \text{ keV} \]
\[ \langle T_{es} \rangle \approx 2 \pm 3 \text{ keV} \]
\[ \langle n_{es} \rangle \approx 2 \pm 4 \times 10^{12} \text{ cm}^{-3} \]

In Figure 3.1 the effect of the level of D-gas puffing recycling on the requirements for D and T pellet refueling is shown. For the example given below the gas puffing rate is \( \lesssim 10^{23} \text{ D}^0 \text{ s}^{-1} \).

The deposition profile of density from the pellets is shown in Figure 3.2. As can be seen the bulk of the particles cross the scrape-off layer.

In Figures 3.3 and 3.4 the T and D particle flows to the divertor due to pellets are shown during a chunk of time in the burn phase. The fluxes are \( \approx 1 \times 10^{22} \text{ T s}^{-1} \), and \( \approx 3 \times 10^{22} \text{ D s}^{-1} \). The rate of burn up of tritium is \( \approx 6 \times 10^{20} \text{ T s}^{-1} \) so the amount of circulating tritium is undesirably high for this example, suggesting that better penetration is required for the tritium pellets. The flux of D and T to the divertor represents about 30% of the flux crossing the separatrix; and, therefore, provided helium confinement is similar to hydrogen confinement it should be possible to exhaust the helium and maintain a low concentration in the plasma. The average temperature and density profiles are shown in Figures 3.5 and 3.6. The profiles of \( T_e, T_i, \) and \( n_e \) during a chunk of time in the burn phase are shown in Figure 3.7, 3.8 and 3.9.
The average $\alpha$-power produced in the plasma and the average losses by conduction and convection, charge exchange, radiation, and to the divertor are:

- $\alpha$-power 215 MW
- Conduction, Convection 29 MW
- Charge exchange 86 MW
- Radiation 39 MW
- Divertor 61 MW

Thus roughly 30% of the power ends up in the divertor. The throat area is approximately $3200 \text{ cm}^2$; consequently, the power density is $\sim 20 \text{ kW cm}^{-2}$. With the aid of tilted targets and field line expansion it should be possible to handle this level of power. While these first computations indicate the levels of particle and heat flow to a bundle divertor, they exhibit some undesirable features. The tritium burn up is low and the charge exchange bombardment of the wall is large with a high scrape-off temperature. A divertor with lower efficiency for particle removal might be more satisfactory.
EFFECT OF DEUTERIUM GAS DIFFUSION ON D&T PELLET FUELING REQUIREMENTS

Figure 3.1
Figure 3.3

TRITIUM FLOW TO DIVERTOR

[Graph showing tritium flow over time with labeled time intervals (0.0 to 200.0 msecs) and particle counts (10^10 to 0.0 particles/msec).]

PELLETS
Figure 3.4
DEUTERIUM FLOW TO DIVERTOR

TIME (MSEC)
Figure 3.5
5. **STUDIES OF MAGNETIC RIPPLE EFFECTS**  
J. A. Rome, R. H. Fowler and J. F. Lyon (ORNL)

Two Coil Versus Four Coil Bundle Divertor Configurations

Fast injected ions prove to be a sensitive probe for evaluating ripple problems in a tokamak. Superthermal ions have large banana widths and few collisions so that they can quickly drift out of a tokamak if the ripple is too large.

Bundle divertors provide a particularly severe ripple since it approaches 100% at the separatrix. However, the ripple decays quickly away from the divertor so that it may not have too bad an effect away from the separatrix in the plasma core. Recent studies at ORNL have led to advanced divertor configurations which can lower the ripple on axis by almost an order of magnitude. In particular, a four coil design has been proposed which significantly lowers the divertor "fringing field" which causes the ripple in the plasma. We are presently engaged in studying the differences between the two coil and four coil designs.

So far, our study has concentrated on following guiding center particles in the rippled configurations. Since this is a tedious and voluminous process, it is fortunate that several general trends emerge.

In a perfectly axisymmetric tokamak, the guiding center orbit is characterized by three constants of motion, $\mu$, $\varepsilon$, and $p_\phi$. If a tiny amount of ripple is introduced, $p_\phi$ is rigorously not conserved; but, in fact, it (or a substitute for it) still is. To see this, we plot $p_\phi$ versus time for each orbit.

Figure 5.1 shows the orbit of a circulating ion in a rippled field. The striking features of the motion are that $p_\phi$ is conserved away from the ripple
but changes its value after each transit of the ripple. However, the long term behavior of $p_\phi$ is periodic. This means that eventually the particle returns to its initial orbit and that a new invariant (constant of the motion) exists. Thus, circulating ions will be contained "forever" in the absence of collisions. The ripple causes some excursions from the axisymmetric orbit but no "walking out." The only net effect of this is to increase the neoclassical step size. On the other hand, trapped ions may be severely affected by the ripple.

There is some question, however, about what part of the ripple most affects the ions. Numerous orbit followings show that the major effect of the ripple occurs due to bouncing off the small maxima in $|B|$ on either side of the large minimum in $|B|$. Furthermore, this bouncing occurs at the banana tips which occur off the equatorial plane. Thus, the effective ripple due to bundle divertors should perhaps be defined as the maximum to average value rather than the peak-to-peak to average value. The former is much smaller. Also, the full 3-D nature of the ripple is crucial and the equatorial plane may not be the most relevant position to measure ripple. To account for these effects, we are working on plots (shown in Fig. 5.2) which show the full 3-D nature of a $|B|$ surface. The orbits have been calculated for $\alpha$-particles in an INTOR scale device to illustrate the effects of 2 and 4 coil divertor configurations. The orbits are shown in Figures 5.3 and 5.4. The INTOR dimensions used were $R = 5.6m$, $a = 1.4m$, $b/a = 1.6$, $B = 5.24T$, $I = 7.9 MA$, $<\beta> = 0.084$, $\beta_p = 3.15$.

The bundle divertor has rectangular coils, either two or four, which are 1 meter high, and 0.8m wide.

In general, the orbit for the 4 coil case jitters around somewhat less than the 2 coil case. Because the loss of $p_\phi$ is due to a resonance between the bounce frequency and the toroidal drift frequency, the direction and
magnitude of the toroidal drift is important. When $|B_T| \sim 6\%$, a minimum forms in $|B|$ in the equilibrium and the toroidal drift direction reverses. This drastically increases the ripple interaction time for some particles and may lead to severe effects. However, even in these cases, the change in $p_\phi$ before and after the entire interaction with the ripple can be small.

The effect of the divertor on the flux surfaces is shown in Figures 5.5 and 5.6, as can be seen the 4 coil divertor does not have such a deleterious effect.
Figure 5.1  The guiding center orbit of a circulating 40 keV proton in ISX-8 in the presence of ripple coils (shown in the top view) to be used for the ripple injection experiment. Even though the orbit undergoes significant radial deflection due to the ripple, the canonical angular momentum $p^\Phi$ is periodic which indicates that there is a new invariant which is conserved. The dashed lines indicate equally spaced poloidal flux surfaces, $I_{RC}$ is the ripple (or bundle divertor) current, and $\xi = v_{11}/v$ and the poloidal flux $\psi_M$ are specified at the orbit starting point.
Figure 5.2 A typical $|B|$ surface (dashed) in the vacuum toroidal field in the presence of the 2-coil bundle divertor. The two maxima referred to in Figure 5.3 which flank the minimum caused by the divertor are clearly visible.
The guiding center orbits of an α-particle in a prototypical reactor (ETF) with a) two-coil and b) four-coil bundle divertors (shown in the top view at the right, $\phi = 0$). The particle in the two coil case is trapped between the field maxima on each side of the bundle divertor. For the four coil case, the field maxima are smaller and do not trap the particle. In each case, the orbit is near the minimum in $|B|$ that characterizes the high $\beta$ equilibrium. Thus, as it moves radially, the direction of $\nabla B$, and hence the toroidal drift direction, may change.
GUIDING CENTER ORBIT

\[ E_0 = 3500.0 \text{ keV} \quad I_p = 7898.0 \text{ kA} \]
\[ M = 4.0 \quad q_s = 3.6 \]
\[ I_{R_c} = 5000.0 \text{ kA} \quad \beta_p = 3.150 \]
\[ \beta = 0.084 \quad \veps = 0.20 \]
\[ B_0 = 5.24 \text{ T} \quad \psi_M = 0.12 \]
\[ R = 6.40 \text{ m} \]
\[ \phi = 270.00 \]
\[ z = 0.00 \text{ m} \]

Figure 5.4 The guiding center orbits of an α-particle in a prototypical reactor (ETF) with a) two-coil and b) four-coil bundle divertors (shown in the top view at the right, \( \phi = 0 \)). The particle in the two coil case is trapped between the field maxima on each side of the bundle divertor. For the four coil case, the field maxima are smaller and do not trap the particle. In each case, the orbit is near the minimum in |B| that characterizes the high \( s \) equilibrium. Thus, as it moves radially, the direction of \( \nabla B \), and hence the toroidal drift direction, may change.
Figure 5.5 Flux surfaces for a two coil divertor, 10 MA turns per coil.
Figure 5.6 Flux surfaces for a four coil divertor, 5 MA turns per coil.
Design studies for the Long Pulse Technology Tokamak (LPTT) at ORNL have examined several bundle divertor concepts which are applicable for tokamaks with superconducting windings. A particularly attractive one from the INTOR point of view is a bundle divertor with normal copper coils which can be "plugged-in" externally to a tokamak with superconducting TF coils. The normal copper conductor allows the use of thinner neutron shielding, which allows smaller divertor coils with lower magnetic field ripple in the plasma. The "plug-in" arrangement is advantageous in a radiation environment due to the ease of maintenance. The paragraphs below describe the LPTT divertor.

This paper discusses the "plug-in" concept which allows for the removal and reinstallation of a normal conducting divertor without disturbing the thermal insulating vacuum around the superconducting toroidal field (TF) coil set. The divertor and its supporting structure is "plugged-in" thru one of nine typical reentrant ports which connect the outer vacuum enclosure to the plasma chamber. A limited structural analysis was performed to verify the structural feasibility of this concept. This analysis showed that safety factors of three on static limit load could be easily accommodated by the divertor support assembly. This design lends itself to a remote maintenance environment and is scalable to larger reactor devices.

The divertor installation is composed of two assemblies. The divertor assembly which consists of the two divertor coil windings and their mutual
case structure is bolted to the **divertor support assembly.** The divertor support assembly reacts the net electro-magnetic forces acting on the divertor assembly, as well as the gravity loading. The net electro-magnetic forces can be thought of as two types.

The **first** is an outward force along the radial centerline of the divertor which is due substantially to the interaction of the divertor and TF coils. This force is carried by the upper and lower compression covers of the divertor support assembly to an outer frame which is also a part of this assembly. This frame is attached to the upper and lower TF coil torque rings by a pair of tension rods located on the centerline of the divertor. These rods are thermally isolated from the liquid helium temperature torque rings by a series of G10-CR fiberglass washers mounted in an assembly which provides a "pin ended" support. This support allows for angular motion resulting from relative motion during cool down of the TF coil structure. The two supports are located on the divertor centerline to minimize system stress during cool down. The outer legs of the TF coils require increased structural stiffness to react the localized divertor imposed forces acting on them to the upper and lower torque rings. The two tension supports are mounted such that they may be removed from the torque rings for inspection and or replacement without removing the vacuum enclosure cover. The divertor support assembly is mounted on several roller assemblies which carry the system weight during installation. A pair of welded bellows on either side of the outer frame allows for misalignment. This system of rollers and bellows also allows the divertor
to move radially inward during cool down of the TF coil set.

The second type of force acting on the divertor causes an overturning moment about the divertor radial centerline. This moment is caused by the interaction of the divertor and poloidal field (PF) coils. The PF coils located within the LPTT plasma chamber are supported by the plasma chamber. These PF coils experience a net moment which acts opposite to and is substantially equal in magnitude to that on the divertor. Four shear pins allow these moments to balance between the divertor assembly and the PF coils thru the plasma chamber structure. The four fittings into which these shear pins mate are installed with the aid of matched tooling such that they provide a generalized interface for future divertor designs.

This design could be used in a remote maintenance environment to allow removal for inspection and or replacement of a normal conducting divertor. The TF coil set geometry must be chosen to provide sufficient room for the divertor to be withdrawn thru the reentrant port.
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Divertor Current Per Coil</td>
<td>$2.16 \times 10^6$ Amp-Turns</td>
</tr>
<tr>
<td>Average Current Density</td>
<td>2400 Amps/cm²</td>
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<tr>
<td>Average Bore of Windings</td>
<td>.66 Meters</td>
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<tr>
<td>Axial Length</td>
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<tr>
<td>Angle Between Coils</td>
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<tr>
<td>Net Radial Force Outward on Divertor Coil Pair ($F_R$)</td>
<td>$1.33 \times 10^6$ Newtons (300000 lbs)</td>
</tr>
<tr>
<td>Net Moment About Radial Axis Through Divertor Coil Pair ($M_R$)</td>
<td>$1.8 \times 10^4$ Newton-Meters (13000 ft-lbs)</td>
</tr>
</tbody>
</table>

**Diagram:**
- Torque Restraint Reaction (TYP 4 PLCS)
- Tension Restraint Reaction (TYP 2 PLCS)
LPTT WARM BUNDLE DIVERTOR
PLUG-IN INSTALLATION CONCEPT

DIVERTOR TENSION RESTRAINT (TYP 2 PL)
BELLOWS
DIVERTOR TORQUE RESTRAINT (TYP 4 PL)
WARM DIVERTOR SUPPORT ASSEMBLY
MODULAR FIRST WALL
POLOIDAL COIL
DIVERTOR GRAVITY SUPPORT
COLD WALL

COLD TF COIL TORQUE STRUCTURE
ACCESSIBLE THERMAL ISOLATION SUPPORT
PLASMA CHAMBER WALL
WARM BUNDLE DIVERTOR ASSEMBLY
PLASMA
DIVERTOR ASSEMBLY ATTACHMENTS
COLD WALL
LPTT WARM BUNDLE DIVERTOR
PLUG-IN INSTALLATION CONCEPT

PLASMA CHAMBER WALL

COLD WALL

WARM BUNDLE DIVERTOR ASSEMBLY

MODULAR FIRST WALL

DIVERTOR ASSEMBLY ATTACHMENTS

DIVERTOR TENSION RESTRAINT (TYP 2 PL)

WARM COMPRESSION COVER

DIVERTOR TOR E

RESTRAINT (TYP PL)

STANDARD PORT

SUPERCONDUCTING TF COIL
LPTT WARM BUNDLE DIVERTOR
PLUG-IN INSTALLATION CONCEPT

vertical port

vertical port

modular first wall

modular first wall

cold wall

cold wall

superconducting tf coil

superconducting tf coil

divertor torque restraint (tYP 4 PL)
divertor torque restraint (tYP 4 PL)

divertor tension restraint (tYP 2 PL)
divertor tension restraint (tYP 2 PL)

warm divertor support assembly

warm divertor support assembly

warm bundle divertor assembly

warm bundle divertor assembly

DIVERTOR ASSEMBLY ATTACHMENTS

DIVERTOR ASSEMBLY ATTACHMENTS
7. **CHOICE OF INTOR SIZE AND FIELD**

The reference design of INTOR has the following parameters:

\[ R_0 \approx 5.0 \, \text{m} \]
\[ a \approx 1.2 \, \text{m} \]
\[ a_s \approx 1.3 \, \text{m} \]
\[ b/a \approx 1.6 \]
\[ B_0 \approx 5 \, \text{T} \]
\[ B_m \approx 10 \, \text{T} \]
\[ B_s \approx 4 \, \text{T} \]

Where \( a_s \) is approximately the separatrix radius for a bundle divertor and \( B_s \) is the toroidal field at the separatrix with the divertor turned off. For these parameters it appears difficult to fit a 2-coil bundle divertor with modest ripple because \( \approx 0.4 \, \text{m} \) is required for neutron shielding and this leads to a divertor coil radius \( \approx 0.5 \, \text{m} \). This large radius coil, given that \( a_s \approx 1.2 \, \text{m} \) leads to high ripple on axis \( > 2\% \).

If a bundle divertor is to be used then a larger value of \( R \) and \( a \) would be desirable. This section summarizes some of the scaling arguments that apply.

a) The field on the plasma axis

\[ B_0 \approx B_m \left( \frac{R_0 - a_s - \Delta}{R_0} \right) \]

Where \( \Delta \) is the distance from the inner edge of the plasma to the inner edge of the toroidal coil \( \Delta \) consists of the "limiter" depth, the first wall thickness, the neutron shield for the toroidal coils and the dewar.
We will take $\Delta = 1.2m$ for the examples below.

b) The field at the separatrix is

$$ B_s = \frac{B_o R_o}{(R_o + a_s)} $$

c) A goal of the INTOR is to produce a certain neutron flux at the wall, this flux is given approximately by

$$ P_{\text{wn}} \propto \frac{a(b/a)^{1/2}}{<nT>^2} $$

d) The limit on $<nT>$ is set by beta, we will assume

$$ \beta_c \propto <nT> \propto \frac{q_o}{q_a^2} \frac{a}{R} \beta_p \frac{a}{R} (1 - \frac{\beta_p a}{R})^{3/2} $$

where

$$ \beta_p \propto \frac{a^2 b/a <nT>}{I^2} = \frac{R}{a} $$

For constant $P_{\text{wn}}$, $b/a$, and $\frac{q_o}{q_a}$

$$ B_o \propto (R/a)^{1/2} \cdot a^{-1/4} $$

$$ I \propto a^{3/4} (\frac{R}{a})^{-1/2} $$

e) For a given divertor the divertor current $I_d$ is

$$ I_d \propto B_s $$

The divertor power $P_d \propto B_s^2$

The field generated at the plasma axis is

$$ B_{od} \propto \frac{I_d}{L_d^2} $$

where $L_d$ is the distance from the plasma axis to the center of the front limbs of the divertor.
The values of these quantities for constant divertor size, $P_{\text{on}}$, $b/a$ and $q_o/q_a$ with $\beta_p = R/a$ are given in Table 7.1, for the INTOR reference device and a somewhat larger device. The larger device offers obvious advantages, the fields are lower, the divertor power is lower and the ripple is lower. Offsetting this is the higher current, but there is more space at the center of the machine for the magnetizing winding. In addition, it is possible to run a similar plasma within the larger vacuum vessel to that used in the reference INTOR design, with lower ripple. Then in turn might permit the use of fewer toroidal field coils, giving better access. Clearly the question of control of the plasma with a greater distance to the poloidal coils would have to be folded in to this optimization.

In summary, far more work is required on optimizing the total tokamak/divertor/poloidal coil/... system.
<table>
<thead>
<tr>
<th></th>
<th>INTOR Reference</th>
<th>Larger INTOR</th>
<th>Larger INTOR Smaller Plasma</th>
</tr>
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<tbody>
<tr>
<td>$R_0$ (m)</td>
<td>5.0</td>
<td>5.6</td>
<td>5.2</td>
</tr>
<tr>
<td>$a$ (m)</td>
<td>1.2</td>
<td>1.5</td>
<td>1.2</td>
</tr>
<tr>
<td>$R/a$</td>
<td>4.17</td>
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<td>4.33</td>
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<td>$a_\omega$ or $a_s$ (m)</td>
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<td>1.7</td>
<td>1.3</td>
</tr>
<tr>
<td>$\Delta$ (m)</td>
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<td>1.2</td>
<td>1.2</td>
</tr>
<tr>
<td>$B_0$ (T)</td>
<td>5</td>
<td>4.5</td>
<td>4.8</td>
</tr>
<tr>
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<td>9.3</td>
</tr>
<tr>
<td>$B_s$ (T)</td>
<td>4</td>
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</tr>
<tr>
<td>$I$ (MA)</td>
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<td>3.9</td>
</tr>
<tr>
<td>$\frac{B_{od}}{B_0}$ (arb)</td>
<td>1</td>
<td>~0.6</td>
<td>~0.4</td>
</tr>
<tr>
<td>$P_0$ (arb)</td>
<td>1</td>
<td>0.72</td>
<td></td>
</tr>
</tbody>
</table>