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EPR-77
A Revised Design for the
Tokamak Experimental Power Reactor*

Fusion Power Program
Argonne National Laboratory
Argonne, Illinois 60439

MASTER

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EPR-77

A Revised Design for the
Tokamak Experimental Power Reactor*

by

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ABSTRACT

A new, preliminary design has been identified for the tokamak experimental power reactor (EPR). The revised EPR design is simpler, more compact, less expensive and has somewhat better performance characteristics than the previous design, yet retains many of the previously developed design concepts. This report summarizes the principle features of the new EPR design, including performance and cost.

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INTRODUCTION

Three design studies⁽¹⁻⁵⁾ for a tokamak Experimental Power Reactor (EPR) were conducted during 1974-1976. The general objective of these design studies was to define the characteristics of a net electric power producing experimental reactor that could operate in the mid- to late-1980s. The designs that evolved were, in general, based upon relatively conservative assumptions about the plasma physics (trapped particle mode energy confinement and low- to moderate- β_t) and upon an attempt to minimize extrapolations beyond present-day technology. The conservative low- to moderate- β_t (3-5%) design basis led to rather large, $R > 6$ m, devices in two of the designs.^(2,4) The third design⁽³⁾ was based upon a relatively high β_t (10%) obtained by shaping the plasma into a highly elongated ($b/a = 3$) Doublet configuration, which resulted in a smaller, $R = 4.5$ m, device.

Recent studies⁽⁶⁻⁹⁾ and earlier work⁽¹⁰⁾ indicate the existence of MHD equilibria with β_t in excess of 10% for circular and D-shaped plasmas with elongations in the range $1 \leq b/a \leq 2$. Although the examination of the MHD stability limits on these high- β equilibria is at an early stage, the work to date suggests that a moderately elongated ($b/a = 1.3 - 1.7$) plasma with $\beta_t \approx 8-10\%$ would be an appropriate basis for an EPR design.

The design concepts that have evolved for the major reactor systems in the previous EPR design studies represent a good starting point for the new EPR design. In addition, a number of improvements have been identified as a result of a re-evaluation of the previous EPR designs and of the continuing EPR design study at ANL.

These considerations have led to the definition of a new design point for the EPR, which is described in this report. This design will be developed in greater detail during this year, and a more complete report will be issued at a future time.

I. OBJECTIVES

The objectives of the ongoing EPR study are to:

- Define the characteristics of a first generation tokamak power reactor that is capable of producing $\sim 30\text{--}50$ MW(e) net electrical power and that would operate in the early 1990s.
- Simplify the design and reduce the cost, relative to the previous EPR design.
- Examine the applicability of the EPR as an engineering and materials test facility.
- Identify the research and development required for an EPR.

II. INNER SHIELD THICKNESS

A prudent tokamak reactor design must be optimized as to the requirements on and utilization of the space on the inner side of the torus from the plasma boundary to the toroidal-field (TF) coil windings. This space is occupied by the scrape-off region, blanket/shield, maintenance clearance space, and the dewar, thermal and magnetic shield, and bobbin of the TF coils. Minimizing this space leads to a higher reactor power for a given maximum toroidal magnetic field strength, a lower ohmic-heating field and better plasma confinement. As shown previously,^(2,11,12) this space is of particular importance in EPR and similar small-size machines. The largest portion of this space, however, is occupied by the blanket/shield region that must satisfy the energy conversion and magnet protection requirements.

The inner blanket/shield design evolved from an extensive parametric study of the conflicting requirements for EPR, including technical and cost considerations. In particular, the study included an in-depth investigation of the radiation effects, local nuclear heating and refrigeration power requirements in the superconducting TF magnets. The study resulted in two conclusions concerning the inner blanket/shield: (1) the composition is an

effective combination of stainless steel and boron carbide in an arrangement similar to that described in reference 2; and (2) the effective inner blanket/shield thickness, Δ_{BS}^i , is 1 m. The parameter Δ_{BS}^i is defined as the distance in midplane from the plasma surface of the first wall to the innermost layer of superconductors where the maximum toroidal magnetic field occurs.

Figure 1 shows some of the important performance parameters for the TF coils as functions of Δ_{BS}^i . These parameters are:

- Neutron fluence, ϕt , and the corresponding reduction in the critical current density, $\Delta J/J_c$, of the NbTi superconductor. At $\Delta_{BS}^i = 1$ m, $\Delta J/J_c$ is $\sim 3\%$ after 4 MW-yr/m² irradiation, which is the first-wall design limit.
- The radiation-induced resistivity, ρ_r , in the copper stabilizer (ρ_{r1} is at 1 MW-yr/m² and ρ_{r2} is at 4 MW-yr/m²). At $\Delta_{BS}^i = 1$ m, $\rho_{r1} = 7 \times 10^{-9}$ Ω -cm, and $\rho_{r2} \cong 3 \times 10^{-8}$ Ω -cm. Radiation damage to the copper and superconductor can be annealed near room temperature.
- D_1 and D_2 are the absorbed doses in the interturn (electrical) and thermal insulations, respectively. Both types are aromatic-based epoxies. At $\Delta_{BS}^i = 1$ m, $D_1 = 7 \times 10^8$ rad, and $D_2 = 4 \times 10^9$ rad after 4 MW-yr/m² irradiation. There is a very serious lack of data on low temperature radiation effects in organic insulators. Extrapolation of available data at high temperatures suggests that aromatic-based epoxies are the most resistant to radiation damage and that they can probably be operated up to a dose of $\sim 10^9 - 10^{10}$ rad. The thermal insulators (located inside the dewar but outside the coil form) receive a much higher dose than the interturn insulators. It should be noted, however, that the performance requirements for thermal insulators are much less severe than those for electrical insulations.
- P_r is the electric power required to run the refrigerators for removal of nuclear energy deposition. At $\Delta_{BS}^i = 1$ m, $P_r = 5.2$ MW(e). This is the maximum instantaneous refrigeration power. The average power over the burn cycle is 2.8 MW(e).

Figure 1. Radiation damage parameters and refrigeration power in the superconducting TF coils as functions of inner clearance, Δ_{BS}^i , from wall to superconductor. All parameters are based on 4.0 MW-yr/m² except ρ_{r1} , which is based on 1 MW-yr/m². Refrigeration power is based on 1.2 MW/m².

III. REACTOR PHYSICS

A. EPR Sizing Studies

The MHD equilibrium and global power balance codes* were used to perform a sizing study for the EPR. Two D-shaped plasmas were considered, with elongations of $b/a = 1.3$ and 1.65 . The $b/a = 1.3$ case was motivated by previous studies,** where it was found that such an elongation was the maximum that could be maintained without requiring stabilization against vertical displacements. There are indications that the $b/a = 1.65$ elongation should maximize the β_t for which the plasma is Mercier-stable.⁽⁶⁾ A range of allowable MHD equilibria was studied for each aspect ratio and plasma elongation by varying the pressure and diamagnetic/paramagnetic ($F = RB_t$) profile and amplitude parameters, with the solutions being constrained to $q(0) = 1$ to avoid kink-mode instabilities. Solutions with $q(\psi_\ell) = 3.0$ were chosen because power reactor performance is known to vary inversely with $q(\psi_\ell)$ in theory and energy confinement has been observed to deteriorate for $q(\psi_\ell) < 3$ in practice. Of these solutions, those with an irregular $q(\psi)$ profile, with the separatrix entering the plasma, with current reversal in any part of the torus, or with sharply spiked current profiles, were rejected, thereby setting an upper limit on the value of β_t for each aspect ratio and plasma elongation.

Equilibria with higher values of β_t than those considered in this work have been obtained using different solution techniques.⁽¹³⁾ However, these higher values of β_t will likely turn out to be MHD unstable.

The plasma power balance equations were solved consistently with the respective MHD solutions. Density and temperature profiles used in the plasma power balance equations were consistent with the pressure profiles in the corresponding MHD solution from which β_t and the plasma current were determined.

Based upon previous work,⁽²⁾ it is estimated that an EPR must produce a maximum power in the vicinity of 600 MW(th) in order to produce a net electrical power of 30-50 MW(e) at a thermal conversion efficiency of $\sim 25\%$ and a duty cycle of $\sim 75\%$. The major radii of reactors with aspect ratios in the range

* See Appendix C of reference 2.

** See Appendix J of reference 2.

$3 \leq A \leq 4$, that would produce 600 MW(th) operating at various values of β_t are shown in Tables 1 and 2 for plasma elongations of $b/a = 1.3$ and 1.65 , respectively. These calculations are based upon a 1 m distance between the conductor on the inner leg and the first-wall ($\Delta_{BS}^1 = 1$ m), a maximum field at the toroidal-field coil of 8 T, an 0.2 m separation between the plasma and the first wall ($\Delta_V = 0.2$ m) and an average electron temperature of 8 keV. The 14-MeV neutron wall load, P_W ; the plasma current, I ; an estimate of the field swing in the ohmic-heating coils, ΔB_{OH} ; the ratio of the energy confinement required for ignition to that predicted by trapped-ion-mode theory, α_{TIM} ; and a judgement as to the credibility of the MHD equilibria based upon the criteria mentioned previously are shown.

A number of interesting features emerge from an examination of Tables 1 and 2. For a given β_t , the major radius of a fixed power EPR is almost independent of aspect ratio over the range considered. The decrease in plasma volume is compensated by the increase in toroidal field, $B_t = B^{TFC} [1 - (\Delta_{BS} + \Delta_V)/R - (1/A)]$, as A increases, so that $P \propto B_t^4 V_P$ remains approximately constant at fixed R . An increase in A , which decreases the engineering complexity, unfortunately also decreases the maximum credible β_t . This maximum credible β_t is larger for $b/a = 1.65$ than for $b/a = 1.3$ for any value of A . For a given value of β_t , a somewhat smaller reactor is possible with $b/a = 1.65$ than with $b/a = 1.3$. However, the $b/a = 1.65$ reactor has a larger plasma current and ohmic-heating coil field swing than the $b/a = 1.3$ reactor operating at the same β_t , which may offset the size advantage. All of the cases considered would achieve ignition with trapped-ion-mode energy confinement; i.e., $\alpha_{TIM} < 1$.

B. Reference Design Parameters and Fallback Positions

The results of the equilibrium parametric study presented in Tables 1 and 2 were used to develop reference design parameters and to evaluate high-field fallback positions. Based upon these results, two designs were selected for the purpose of comparing the performance resulting from high β_t and high field operating conditions. Operating parameters for these two designs are given in Tables 3 and 4 for $B^{TFC} = 8, 9$ and 10 T, with the toroidal-field coil design based upon a maximum strain in the conductor of $\epsilon = 0.2\%$. These two designs have about the same maximum credible β_t and have identical power output for identical values of β_t and B^{TFC} . The $b/a = 1.3$ design has a slightly lower plasma current and ohmic-heating coil field swing than the $b/a = 1.65$ design.

Table 1. Sizing study for an EPR with plasma elongation $b/a = 1.3$ ($d = 0.25$) that produces $P_T \sim 600$ MW(th). ($q(0) = 1$, $q(\psi_\ell) = 3.0$, $B^{TFC} = 8$ T, $\langle T_e \rangle = 8$ keV, $\Delta_{BS} = 1$ m, $\Delta_V = 0.2$ m)

	<u>A</u>	<u>R_O (m)</u>	<u>P_w (MW/m²)</u>	<u>I (MA)</u>	<u>ΔB_{OH} (T)</u>	<u>α_{TIM}</u>	<u>MHD</u>
$\beta_t = 6\%$	3.0	6.1	0.8	8.3	5.0	0.4	✓
	3.5	6.0	0.9	6.5	3.6	0.4	✓
	4.0	6.0	1.0	5.4	2.8	0.3	✓
$\beta_t = 8\%$	3.0	5.5	1.0	7.5	5.8	0.5	✓
	3.5	5.3	1.1	5.8	4.2	0.5	✓
	4.0	5.3	1.3	4.8	3.2	0.4	?
$\beta_t = 10\%$	3.0	5.0	1.1	6.8	6.7	0.7	✓
	3.5	4.9	1.4	5.4	4.8	0.6	?
	4.0	4.9	1.6	4.6	3.7	0.5	X
$\beta_t = 12\%$	3.0	4.7	1.3	6.4	7.6	0.8	?
	3.5	4.6	1.5	5.1	5.5	0.7	X
	4.0	No MHD Solution					

Table 2. Sizing study for an EPR with plasma elongation $b/a = 1.65$ ($d = 0.5$) that produces $P_T \sim 600$ MW(th). ($q(0) = 1$, $q(\psi_\ell) = 3.0$, $B^{TFC} = 8$ T, $\langle T_e \rangle = 8$ keV, $\Delta_{BS} = 1$ m, $\Delta_V = 0.2$ m)

	A	R_O (m)	P_w (MW/m ²)	I (MA)	ΔB_{OH} (T)	α_{TIM}	MHD
$\beta_t = 6\%$	3.0	5.9	0.8	11.7	6.9	0.4	✓
	3.5	5.8	0.9	9.1	4.9	0.3	✓
	4.0	5.7	1.0	7.1	3.7	0.3	✓
$\beta_t = 8\%$	3.0	5.2	0.9	10.1	8.2	0.6	✓
	3.5	5.1	1.1	7.7	5.7	0.5	✓
	4.0	5.1	1.2	6.2	4.2	0.4	✓
$\beta_t = 10\%$	3.0	4.8	1.1	9.0	9.2	0.7	✓
	3.5	4.7	1.3	6.9	6.3	0.6	✓
	4.0	4.7	1.5	5.6	4.7	0.5	?
$\beta_t = 12\%$	3.0	4.5	1.2	8.2	10.2	0.8	✓
	3.5	4.4	1.4	6.3	7.0	0.7	?
	4.0	4.4	1.6	5.2	5.2	0.6	X
$\beta_t = 14\%$	3.0	4.3	1.3	7.7	11.2	0.9	?
	3.5	4.2	1.6	6.0	7.7	0.8	X
	4.0	No MHD Solution					
$\beta_t = 16\%$	3.0	4.1	1.4	7.2	12.5	1.1	X
	3.5	No MHD Solution					
	4.0	No MHD Solution					

Table 3. Trade-off between high- β_t and high field for EPR design No. 1 with $R = 5$ m, $A = 3$, $b/a = 1.3$ ($d = 0.25$). ($q(0) = 1$, $q(\psi_g) = 3.0$, $\Delta_{BS} = 1$ m, $\Delta_V = 0.2$ m, $\langle T_e \rangle = 8$ keV).

	B^{TFC} (T)	P_T (MW)	P_w (MW/m ²)	I_P (MA)	ΔB_{OH} (T)	α_{TIM}	MHD
$\beta_t = 6\%$	8	220	0.4	6.2	6.2	1.7	✓
	9	350	0.7	7.0	7.6	0.8	✓
	10	540	1.0	7.8	9.5	0.4	✓
$\beta_t = 8\%$	8	385	0.7	6.5	6.4	1.0	✓
	9	620	1.2	7.3	7.9	0.5	✓
	10	945	1.8	8.1	9.9	0.3	✓
$\beta_t = 10\%$	8	595	1.1	6.7	6.7	0.7	✓
	9	950	1.8	7.6	8.3	0.3	✓
	10	1450	2.8	8.4	10.4	0.2	✓
$\beta_t = 12\%$	8	835	1.6	7.0	7.0	0.5	?
	9	1340	2.6	7.9	8.7	0.3	?
	10	2040	3.9	8.8	10.8	0.1	?

Table 4. Trade-off between high- β_t and high field for EPR design No. 2 with $R = 4.7$ m, $A = 3.5$, $b/a = 1.65$ ($d = 0.5$). ($q(0) = 1$, $q(\psi_\rho) = 3.0$, $\Delta_{BS} = 1$ m, $\Delta_V = 0.2$ m, $\langle T_e \rangle = 8$ keV).

	B^{TFC} (T)	P_T (MW)	P_w (MW/m ²)	I_p (MA)	ΔB_{OH} (T)	α_{TIM}	MHD
$\beta_t = 6\%$	8	220	0.5	6.7	6.1	1.4	✓
	9	355	0.7	7.5	7.5	0.7	✓
	10	540	1.2	8.3	9.3	0.4	✓
$\beta_t = 8\%$	8	390	0.8	6.8	6.2	0.8	✓
	9	625	1.3	7.6	7.6	0.4	✓
	10	950	2.0	8.5	9.4	0.2	✓
$\beta_t = 10\%$	8	600	1.3	6.9	6.3	0.6	✓
	9	960	2.1	7.8	7.8	0.3	✓
	10	1460	3.1	8.6	9.6	0.2	✓
$\beta_t = 12\%$	8	840	1.8	7.0	6.4	0.4	?
	9	1340	2.9	7.9	7.9	0.2	?
	10	2050	4.4	8.8	9.8	0.1	?

Based on the results presented in Tables 1 through 4 and stability considerations, the 1977 EPR reference operating conditions were established at $\beta_t = 8\%$ and $B^{TFC} = 9$ T at 4.2 K for both geometries presented in Tables 3 and 4. If higher values of β_t are achievable, the EPR could operate with a higher power output at 9 T or at the same power output at reduced field. If, on the other hand, β_t is limited to less than 8% by MHD stability, the EPR could operate at fields up to about 10 T by pumping down to lower temperatures in order to compensate for the effect of the reduction in β_t on the power ($P \propto \beta_t^2 B^4$).

For the purpose of developing a representative set of parameters characterizing the EPR, Reference Design No. 1 ($R = 5$ m, $A = 3.0$, $b/a = 1.3$) will be the basis of the remaining design parameters presented in this report. The ultimate choice between designs No. 1 and No. 2 will be made after the evaluation of MHD stability limits on β_t . There is no evidence of instability due to localized ideal or resistive interchange modes (Mercier criterion) in Design No. 1 for β_t up to 10%.

C. Burn Cycle Performance

A series of burn cycle calculations, in which the dynamics of the plasma, the plasma driving system and the plasma heating system are simulated dynamically,* have been performed. The nominal sequence of events is: (1) the plasma is initiated with a 10 ms voltage pulse provided by a set of initiation-trimming coils; (2) the current in the OH coils is reversed through a bank of homopolar generators in $\Delta t_{OH} = 2$ s; (3) then 40 MW of supplemental heating power (neutral beam or rf) is injected in the interval $1 \text{ s} \lesssim t \lesssim 7.5 \text{ s}$ to heat the plasma to $\langle T_e \rangle \sim 8$ keV; (4) the plasma burns until the accumulation of wall-sputtered impurity (Be-coated first wall) and helium quenches it; (5) the OH current is reversed to terminate the burn; and (6) the vacuum chamber is evacuated and replenished in 15 s to prepare for the next burn pulse.

The length of the burn pulse and the net electrical power depend upon the energy confinement time and the rate of impurity accumulation. Using an upgraded version of the plasma-wall interaction model of reference 2, the net electrical power and the burn pulse length were calculated as a function of the energy confinement (measured relative to the trapped-ion-mode prescription)

* See reference 2 for a description of the computational model.

and the plasma edge temperature, which affects the sputtering rate. As shown in Figure 2, the EPR power objectives can be met if energy confinement is at least as good as predicted by the trapped-particle theory ($\alpha_{TIM} \geq 1$). Note that the P_{net} shown in Figure 2 must be reduced by the duty cycle factor to obtain the cycle-averaged net power. Somewhat better power performance can be achieved with a modest amount of supplemental heating to extend the burn pulse length. No means of impurity control other than coating the first wall with beryllium were assumed in these calculations.

The basis of the net power computation is indicated schematically in Figure 3, where the net energy flow during a nominal burn cycle ($\alpha_{TIM} = 2$, $\Delta t_{burn} = 65$ s) is summarized. The net electric power averaged over the 80 s burn cycle (65 s burn + 15 s down) is 39 MW(e).

D. Plasma Driving, Heating and Initiation System Requirements

Requirements on the plasma driving, heating and initiation systems are obtained from the burn cycle dynamics simulation and from certain auxiliary calculations (e.g. beam penetration, plasma breakdown). These requirements are summarized in Table 5.

IV. MAGNET SYSTEMS

The magnet systems consist of the toroidal-field (TF) coils, the ohmic-heating (OH) and equilibrium-field (EF) coils and the initiation and trimming (IT) coils. For economy of operation, the TF, OH and EF coils are superconducting; they all use an NbTi conductor with copper stabilizer. There are sixteen TF coils of the constant-tension shape. The EF and OH coils are located outside the TF coils to facilitate assembly, disassembly and maintenance of the coils, first wall, blanket and shield. The coil locations are shown in Figure 4. The TF coils are surrounded by high-purity aluminum to shield them from the pulsed field of the EF coils and the plasma.

Figure 2. EPR Burn Cycle Duration and Power Performance

Figure 3. Net Energy Flow in Reference Burn Cycle

Table 5. Plasma Driving and Heating System Requirements

Ohmic Heating System

Volt-second to plasma	
startup	45 V•s
burn	10 V•s
total	55 V•s
Current reversal time, Δt_{OH}	2 s
Peak field, B_{OH}	4.0 T
Maximum field rise, $\dot{B}_{OH}$	4.5 T/s
Maximum voltage, V_{OH}	25 kV
Maximum current, I_{OH}	56 kA
Maximum energy transferred during startup, U_{OH}	430 MJ

Equilibrium Field System

Volt-second to plasma	31 V•s
Maximum voltage, V_{EF}	9.4 kV
Maximum current, I_{EF}	80 kA
Maximum energy transferred, U_{EF}	732 MJ
Maximum reactive power, P_{EF}	756 MVA

Plasma Heating System

Power to plasma, P_B	40 MW
Heating time, Δt_B	6.5 s
Neutral beam energy, U_B	
normal injection	180 keV

Plasma Initiation System

Breakdown voltage, V_{LOOP}	450 V
Breakdown voltage pulse length, Δt_{BD}	10 ms

Figure 4. Schematic Views of EPR.

A. Toroidal-Field Coils

Parameters for the TF coils are shown in Table 6. The peak field from the TF coils is 9 T. This high field for NbTi requires a large amount of superconductor. The amount of superconductor could be cut in half by operating at a temperature of 3.6 K rather than 4.2 K. Tentatively, 4.2 K operation at 9 T has been chosen for the reference design. The backup design position of raising the peak operating field to 10 T would be accomplished by lowering the temperature.

The allowable strain of 0.2% in the TF coils is set to limit the strain-induced resistivity of the stabilizing copper in the coils. The NbTi could stand a higher strain without degrading. A relatively small quantity of stainless steel wound with the conductor is adequate to keep the strain to 0.2% for fields of 9 T and below. For a peak field of 10 T, the strain in the copper can still be held to 0.2% by winding the copper in precompression and the stainless steel in pretension. Otherwise, the strain would rise to 0.25%.

The thickness of the support cylinder was chosen to give a stress in the stainless steel of 50,000 psi for a peak field of 9 T. For a peak field of 10 T, the stress would be 62,000 psi, which is still a safe value relative to the yield stress of 100,000 psi.

The overall size of the TF coils is determined by the size of the vacuum chamber, blanket and shield that lie inside the TF coils. The inner and outer radii are respectively 1.98 m and 8.82 m. The stored energy is 11.4 GJ for 9 T peak field. The coil size and shape and the amount of NbTi, copper and stainless steel required, as well as the other parameters in Table 6, were determined using the magnet model code recently developed at ANL.⁽¹⁴⁾

Cryogenic stability is essential for a magnet system of this size. For a conductor of the type described below, the average current density in the copper should be 5.1 kA/cm². Because most of the resistivity of the copper is magnetoresistivity and radiation-induced resistivity, both of which vary across the thickness of the coil, the copper can be graded, with more copper on the inner turns and less on the outer turns where the radiation level and magnetic field are lower. The NbTi is also graded; less is required in the

Table 6. Toroidal Field Coil Parameters

Number of coils	16
Coil shape	Pure-tension D
Coil thickness	0.31 m
Mid-radius, inner leg	1.98 m
Mid-radius, outer leg	8.82 m
Coil height	10.73 m
Mean turn length	28.52 m
Minimum radius of curvature	1.64 m
Allowable strain on conductor	0.2%
Operating temperature	4.2 K (3.6 K)
Peak field	9 T (10 T)
Field at plasma	3.84 T
Total amp-turns	96 MA-turns
Average current density over Cu	5090 A/cm ²
Operating current	40 kA
Turns per coil	38 x 4
Inductance	57 Henry
Stored energy	11.4 CJ
Weight of copper	480 Mg
Weight of stainless steel	160 Mg
Weight of NbTi	38 Mg

outer turns where the field is lower and the critical current density higher. Cooling is by pool-boiling helium at 4.2 K or 3.6 K as required.

A 40 kA sheet conductor is envisioned, consisting of a sheet of copper stabilizer with many superconducting composite wires stranded around it and soldered to it. Stainless steel reinforcement is wound with the sheet conductor into four pancakes per coil.

Such a conductor is inherently a dc conductor and would exhibit large eddy current losses and other ac losses if it were subjected directly to the pulsed field of the EF coils and plasma. Therefore, the TF coils are enclosed by high-purity aluminum field shields, operating at low temperatures (4.2 K to 20 K) to further improve their electrical conductivity. The aluminum shield system requires development. The alternative is to use a conductor in the TF coils that is designed to minimize ac losses, but such a conductor would probably have poorer mechanical properties and would also require substantial development.

The TF coils must be operated in series; otherwise, they face the possibility of carrying unequal currents, which will produce large bending moments and out-of-plane forces in the coils. However, with series operation an excessively large voltage may develop upon discharge unless suitable protection circuitry is available. Such a protection scheme could consist of dump resistors between neighboring coils, which are shunted during normal operation but are in series with the coils during discharge.

B. Ohmic-Heating and Equilibrium-Field Coils

The OH and EF coils are located outside the TF coils (see Figure 4) to facilitate maintenance and assembly/disassembly operations. Parameters for the coils are shown in Table 7.

1. OH Coils

The OH coils consist of a central solenoid 10.4 m high plus ten other coaxial coils. When energized they provide a large flux (up to 55 Weber) linking the plasma, but with a low field over the regions of the plasma ($< 10^{-3}$ T) and TF coils (< 0.1 T). The OH coil locations and currents to

Table 7. Poloidal Field Coil Parameters

	<u>OH Coil</u>	<u>EF Coil</u>
Operating temperature	4.2 K	4.2 K
Maximum operating current	64 kA	80 kA
Current density	2110 A/cm ²	2640 A/cm ²
Total amp-turns	38 MA-turns	18 MA-turns
Maximum volt-seconds	55 V·s	31 V·s
Maximum field at conductor	4.0 T	
Maximum rate of change of field	4.5 T/s	
Field at plasma	< 10 ⁻³ T	0.48 T
Field on TF coil	< 0.07 T	< 0.7 T
Inductance	0.33 Henry	0.25 Henry
Coupling coefficient to plasma	-0.291	-0.266
Energy transferred	534 MJ	732 MJ
Inductive energy: OH + EF + Plasma	1384 MJ	

give these low fields while providing the required flux were found using the magnet design program GFUN.

In operation, the coils are energized to give their full negative flux before the plasma current is to be excited. The flux is then reversed, inducing a current in the plasma. Throughout the burn cycle the flux is further increased to compensate for resistive losses in the plasma.

2. EF Coils

The EF coils produce a field that holds the plasma in stable equilibrium with the desired position and shape. The 14 EF coils in Figure 4 are inductively

decoupled from the OH coils. The currents in the EF coils are determined with the PPPL free-boundary MHD equilibrium code to provide the required vertical field in the plasma, once the locations are fixed by engineering constraints. The EF coils produce a pulsed field of up to 0.7 T at the TF coil locations; the TF coils are shielded from such a high field by means of the aluminum shield described previously.

3. Conductor

The EF and OH coils must be cryostable. The conductor is a 40 kA fully transposed cable of copper and composite strands. Equivalent coils from above and below the midplane would be operated in parallel to provide for an 80 kA operational current.

C. Initiation and Trimming Coils

A system of copper coils within the blanket and shield is planned to produce the flux change necessary to initiate the plasma and provide the fields necessary for trimming plasma position. A variety of coil geometries are being studied, with the aim of making this system compatible with the magnetic and radiation environments.

V. PLASMA HEATING SYSTEMS

A number of different alternatives are being considered for heating the plasma to ignition. These methods must be capable of producing 40 MW of power for about 6 to 8 seconds to heat the plasma. Any system used in an EPR must be relatively simple, rugged, efficient and flexible enough to follow changes in density, temperature and major and minor radii. The radiation environment also imposes severe constraints. The chief alternatives are neutral beam heating and radio frequency (rf) heating. Neutral beams are presently considered to be the primary option.

A. Neutral Beam Injector Design

The analytical methods developed for last year's EPR studies⁽²⁾ are being used in a continuing study of the neutral beam injector system for EPR. As in last year's EPR study, three reference designs are being considered. Reference

Designs 1 and 2 are based on the use of positive deuterium ion sources and assume minor and major extrapolations, respectively, beyond the state of the art required for TFTR ion sources. Reference Design 3 is based on the use of hypothetical direct-extraction D^+ ion sources with dc current densities and gas efficiencies comparable to those assumed for Reference Designs 1 and 2. Given the current level of effort, it is not likely that a D^+ source with the assumed characteristics will be developed in time to be of use for EPR. The major emphasis at this time is, therefore, on Reference Designs 1 and 2.

Reference Designs 1 and 2 both call for twelve injectors, with two beam lines per injector, to provide 40 MW of 180 keV D^0 beam. Each beam line will have a positive ion source and accelerator, followed by a double-focusing bending magnet, a neutralizer for the atomic ions and a second bending magnet to remove atomic ions from the beam emerging from the neutralizer. Direct energy convertors will be used to recover energy from the molecular and atomic ions removed by the first and second separator magnets. Thermal energy recovery will also be used to increase the overall injector efficiency. The two neutral beams from each injector will converge toward an 0.75-m diameter aperture in the first wall of the torus at angles of 10° above and 10° below the horizontal median plane of the torus.

In Table 8, some preliminary results of a study of a 40 MW neutral injector system are compared with results obtained in last year's study⁽²⁾ of a 60 MW system. It may be noted that going from 60 MW to 40 MW results in large reductions in beam power requirements, injector gas loads and net power input; these reductions are accompanied by significant increases in electrical and overall power efficiencies. These efficiencies are functions of Chamber I and Chamber II gas loads and pumping speeds. [Each injector is partitioned into two chambers. Chamber I contains the first bending magnets (one for each beam line) direct-convertors and the first half of each neutralizer; Chamber II contains the last half of each neutralizer, the second bending magnets and direct-convertors.]

For the 40 MW cases, the overall power efficiency is reduced by 0.9% in Reference Design 1 and by a negligible amount in Reference Design 2 when the

Table 8. Characteristics of 40 MW and 60 MW EPR Neutral Beam Injector Systems

	<u>Reference Design 1</u>		<u>Reference Design 2</u>	
Atomic ion	D^+		D^+	
Target for $D^+ \rightarrow D^0$	D_2 gas		D_2 gas	
Beam composition (D^+ , D_2^+ , D_3^+)	(0.75, 0.18, 0.07)		(0.95, 0.03, 0.02)	
Neutral beam energy (keV)	180		180	
Neutral beam power to plasma (MW)	<u>40</u>	<u>60</u>	<u>40</u>	<u>60</u>
Number of injectors	12	12	12	12
Sources/injector	2	2	2	2
Ion beam current density (A/cm ²)	0.135	0.135	0.175	0.175
Gas efficiency	0.45	0.45	0.50	0.50
Ion beam power (MW)	261	441	202	338
Gas load/injector (Torr-ℓ/s)	61.3	110.0	31.0	57.0
Pumping speeds, Chambers I (II) (ℓ/s)	$2(2) \times 10^6$	$3(2) \times 10^6$	$2(2) \times 10^6$	$2(2) \times 10^6$
Cryopanel area, Chamber I (m ²)	40	60	40	40
Cryopanel area, Chamber II (m ²)	40	40	40	40
Direct conversion efficiency	0.85	0.85	0.85	0.85
Thermal conversion efficiency	0.25	0.30	0.25	0.30
Electrical power efficiency	0.34	0.29	0.45	0.41
Overall power efficiency	0.37	0.34	0.48	0.45
Net power input to injectors (MW)	119	207	90	145

Chamber I and II pumping speeds are reduced to 1.5×10^6 l/s; the total required cryopanel area is reduced from 80 m² to 60 m². The electrical power efficiencies are reduced by 1% for the 40 MW Reference 1 design and by a negligible amount for the 40 MW Reference 2 design when the pumping speeds are reduced from 2×10^6 l/s to 1.5×10^6 l/s.

B. RF Heating

A system to deliver 40 MW of ion-cyclotron heating (ICRH) using ridged waveguides is under study. The guide, shown in Figure 5, should be capable of delivering 2.5 MW to the plasma at 58.5 MHz. Sixteen of these would be required to deliver 40 MW. The waveguide could be coupled to a 9-inch coaxial cable inside the outer blanket. The design of dc breaks and windows is also under study.

The lower-hybrid heating (LHRH) system plan would be similar to that proposed in the previous design.⁽²⁾ A phased array, the Lallia-Brambilla "Grill",⁽¹⁵⁾ of reduced height, WR 510 waveguide is under study. An array of guides, four by ten (in the toroidal direction), is capable of delivering 8 MW to the plasma. A total of five of these would be required to deliver 40 MW to the plasma.

A system of heating using electron-cyclotron heating (ECRH) is also being studied. Tubes with CW power levels of 100 kW are being developed, and systems capable of using these tubes are being designed.

VI. VACUUM SYSTEMS

A. Toroidal Vacuum System

Using a computer code developed to calculate fuel cycling and vacuum system parameters for D-T burning tokamaks (TCODE), the fuel recycle requirements for the EPR were evaluated as a function of duty factor, plasma parameters and duct dimensions. Because it is desirable to minimize the number of penetrations through the first wall, a decision was made to place the vacuum pumps and ducts on the neutral beam lines. In the present design this permits

Figure 5. Ridged Waveguide

twelve pumping stations to be active at any given time. For purposes of this study, the entire vacuum duct was conservatively taken as circular, with a length of 6 m.

The pump capacity requirements show that for an 80% duty factor and twelve pumps, a pump capacity of 64,500 torr-ℓ is required, assuming a four-hour regeneration time. This translates to a pump having a speed of 80,000 ℓ/s. A list of the vacuum system parameters for this design is given in Table 9. The data show that conditions necessary to have a regeneration period of four hours result in a minimum duct diameter of 0.73 m.

B. Neutral Beam Vacuum System

Radiation streaming into the beam injectors has cast some doubt on the practical as well as economical use of cryopumping in the neutral beam lines. Operating experience with a zirconium-aluminum pump on an ultra-high vacuum system in the vicinity of a fission-reactor core⁽¹⁶⁾ supports the possibility of using this pumping method for the neutral beams. Recent tests with a panel containing 1500 cm² of active area resulted in pumping speeds for hydrogen of 15,000 ℓ/s and a storage capacity in excess of 19,000 torr-ℓ. Since temperatures of 200°C or more are needed for most efficient pumping, the zirconium-aluminum pump would, in fact, be aided by the radiation heating due to the neutron flux, thus requiring less auxiliary power to reach its most efficient operating temperature. Neutral beam vacuum system parameters are given in Table 9, for both cryosorption and zirconium-aluminum panel designs.

VII. ENERGY STORAGE AND TRANSFER

The energy storage and transfer (EST) system is a scaled down version of the EST system developed in the previous EPR design.⁽²⁾ This EST system consists of a central energy storage inductor (ESI), rectifiers to transfer energy between the ESI and the OH coil, EF coil and neutral beam (rf) systems, and a rectifier to transfer makeup energy from a power substation into the ESI. A separate inertial energy storage unit, consisting of radially-stacked homopolar generators, is used to transfer inductive energy into and out of the OH coil system. The EST system is depicted in Figure 6 and summarized in Table 10.

Table 9. Vacuum Systems

	<u>Toroidal</u>
Volume	450 m ³
Surface area	430 m ²
Dwell time	15 s
Post-burn pressure (500°C)	8.3 x 10 ⁻³ torr
Pre-burn pressure (500°C)	1.0 x 10 ⁻⁵ torr
Post-burn gas load	3.7 x 10 ³ torr-l
Pumping speed	1.9 x 10 ⁵ l/s
Number of cryosorption pumps	12 (24 with regeneration)
Duct conductance	2.5 x 10 ⁴ l/s
Minimum duct diameter	0.73 m
Pump rated speed	8.0 x 10 ⁴ l/s

	<u>Neutral Beam</u> (per injector)
Gas load	61.3 Torr-l/s
Volume of injector	300 m ³
Surface area of injector	220 m ²
Effective pumping speed	
Chamber I	2 x 10 ⁶ l/s
Chamber II	2 x 10 ⁶ l/s
Pumping panel area	
Cryosorption panels	40 m ² /chamber
Zr-Al panels	20 m ² /chamber
Zr-Al panel power	
Activation at 700°C	1.4 MW
Operation at 400°C	0.3 MW

Figure 6. Energy Transfer and Storage System

Table 10. Energy Storage and Transfer Systems - Maximum Ratings

Ohmic Heating Systems

Homopolar Generators

No. of generators in series	8
No. of drums per generator	6
Total energy transfer (MJ)	534
Peak power (MW)	840
Peak voltage (kv)	30
Peak current (kA)	56
Equivalent capacitance (f)	1.188

Rectifier System

Type	3 $\emptyset$ Inductor Converter Bridge
Energy transfer (MJ)	350
Peak power (MW)	56
Peak current (kA)	80
Peak voltage (kv)	0.7
Switching frequency (Hz)	100

Equilibrium Field System

Type	3 $\emptyset$ Inductor-Converter Bridge
Energy transfer (MJ)	732
Peak power (MW)	226
Peak current (kA)	80
Peak voltage (kv)	13.5
Switching frequency (Hz)	100

Neutral Beam System

Type	SCR, DC/AC/DC @ 10 kHz
Energy transfer (MJ)	770
Voltage (kv)	180
Power (MW)	120

Central Energy Storage Inductor

Type	Superconductive ring dipole inductor
Energy stored (MJ)	2525
Energy transfer (MJ)	1900
Peak current (kA)	80
Peak power (MW)	350
Average power from utility grid or power plant (MW)	13.3

A. Ohmic Heating System

The OH coil stores 430 MJ at the start of the burn cycle, operates at 56,000 A and reverses in two seconds. If the OH coil is resonated with a capacitor of 1.312 f, field reversal would occur in two seconds with a peak voltage of 24,120 volts developing across the coil terminals. A counter cyclonic generator (CCG) type homopolar generator⁽¹⁷⁾ is envisioned for this purpose.

The conceptual design for the CCG is based on NbTi coils with a magnetic yoke. The drum design, brushes and dynamic design are the same as those described in reference 2. The usable field region allows drums of length 0.55 m to be radially stacked in a radial region from 0.55 m to 0.85 m. Type 17-4 stainless steel is the material envisioned for the conducting cylinders. The density of type 17-4 stainless steel is 7800 kg/m³. The thickness of each drum is adjusted so that the surface velocity of the cylinder reaches 236 m/s for 56,000 amperes of OH current. Table 11 lists the values for radial thickness, induced voltage, effective capacitance and stored energy for each of the conducting cylinders in a single array between field coils.

A single CCG module will develop 3750 volts, have an effective capacitance of 9.507 farads and store an energy of 66.8 MJ. A total of six conducting cylinders are stacked in each module. Eight modules in series will develop 30,000 V and will have a minimum capacitance of 1.188 farads. The minimum field reversal time in the absence of stray inductances and circuit losses would be 1.79 s.

The CCG is disconnected from the OH coil during the burn. The OH coil current and $\dot{B}$ are maintained with a separate power supply driven from a central energy storage inductor (CSI) which is described later. An actively controlled inductor-converter of the type described for the EF coils is used for this application. A total of 350 MJ is taken from the CSI during the burn. The inductor-converter will provide a peak voltage of 700 V and will be reversible.

B. Equilibrium Field System

The equilibrium-field coil is designed to operate at 80 kA and to store a maximum of 732 MJ. The peak EF coil voltage is 9.4 kV at about 1.25 s into

Table 11. Design Values for the CCG Homopolar Generation Module

Cylinder No.	Inner Radius m	$\langle B l \rangle^*$ T-m	Radial Thickness m	Induced Voltage V	Effective Capacitance f	Stored Energy MJ
Inner Insulator	0.55	-	0.019	-	-	-
1.	0.569	2.4	0.0236	566.4	62.94	10.1
2.	0.6147	2.48	0.0226	585.3	60.91	10.4
3.	0.6594	2.57	0.0218	606.5	58.78	10.8
4.	0.7032	2.68	0.0214	632.5	56.37	11.3
5.	0.7467	2.81	0.0211	663.2	53.76	11.8
6.	0.7898	2.95	0.0209	696.2	51.2	12.4
Outer Insulator	0.8328					
Total				3750.1	9.51 f	66.8

* integral value of the radial flux over the length of the conducting cylinder.

the startup phase. The voltage as a function of time has been computed. An actively controlled version of an inductor-converter bridge^(18,19) driven from a central superconducting energy storage inductor is selected as the power supply and energy storage choice.

A circuit diagram for a 3-phase inductor-converter bridge is shown in Figure 7. The EF coil is represented by L_E and the central energy storage coil is represented by L_S . The capacitors in the circuit are relatively small in the sense that they only store a small fraction of the total magnetic energy in the circuit. The left-hand (LH) bridge commutates the L_S current through the three capacitors so as to create cyclic voltages on each capacitor, which are separated 120° in phase from each other. The right-hand (RH) bridge is switched in the same sequence but at a different relative time from the LH bridge. The current in L_E grossly affects the voltage waveshape on the capacitors, especially if the current in L_E exceeds the current in L_S .

The peak voltage required for normal charging of the EF coil is 9400 V. The inductor-converter bridge is designed to generate 13,500 V across the L_E coil as an added control margin. The margin is only 3100 V, or 10,900 A/s increase in EF coil rate of charge, at the highest calculated EF coil voltage for the anticipated startup scenario. At other times in the anticipated startup scenario, the margin is much larger, allowing up to 45,000 A/s.

Typical SCR parameters for this application are listed in Table 12. Each capacitor has a design value of 8500 V, 0.025 farads and 737 kJ. At a switching time of 2.4 ms a voltage of 7680 V is generated across each capacitor. An average value of 9400 V is generated across L_E when the angle ϕ , by which the RH bridge leads the LH bridge, is 46° . Advancing ϕ to nearly 90° generates 13,500 V across L_E . Later in the cycle, some droop occurs in the average voltage as a function of phase angle. This is not too serious since the highest voltage demanded later in the startup scenario is only 3300 V. Even if the storage coil and EF coil current were equal, the bridge can be operated with a phase angle, ϕ , of 20° and still match the design startup scenario, and in fact, have 6223 V margin to increase over the design point.

Figure 7. Inductor-Converter Bridge Circuit Diagram for Equilibrium-Field Coil System

Table 12. SCR Design Parameters for EF Coil Inverter-Converter Network Design

RMS Peak Current	1600 A
Current Handling Limit at 120° Conduction Angle	1400 A
Design Current per SCR	1050 A
Peak Inverse Voltage (Repetitive)	1500 V
Design Inverse Voltage	1000 V
Turn-Off Time	100 to 120 μ s
di/dt Rating	800 A/ μ s
dv/dt Rating	200 V/ μ s
Voltage Drop at Maximum Current Conduction	1.75 V

C. Neutral Beam Power Supply

The design of the neutral beam injector energy transfer system is composed of two major components; a saturated time-delay transformer (STDT),⁽²⁰⁾ which uses the saturation effects of magnetic cores to act as a current surge limiter, and a high frequency, polyphase-controlled rectifier using SCR switches to enable rapid de-energization of the beam in periods of less than 100 μ s. A circuit diagram is shown in Figure 8.

The injector power supply extracts energy from the same central energy storage inductor that the EF coil power supply uses. The peak power demand is never put on the utility grid. An inductor-converter similar to that described for the EF coil system drives a high frequency, polyphase inverter. A 10 kHz voltage is developed in a summing transformer, filtered and subsequently rectified. The output lead is connected to the injector through an STDT. In the event of a spark, the STDT prevents energy transfer into the spark and, if necessary, the polyphase rectifier can be phased back quickly

Figure 8. HV Power Supply for NBI Block Diagram

enough to remove power from the injector in the event that the discharge path has not cleared in a time beyond which the STDT is unable to absorb the energy.

The neutral beam injector system provides 40 MW(e) to the plasma for 6.5 s during startup. This requires an input power to the injector system of 284 MW(e). Direct energy recovery of the molecular fraction of the beam provides 165 MW(e) recirculating power to the injector systems (see Figure 3). Thus, 119 MW(e) additional power must be supplied to the injector system from the ESI.

D. Central Energy Storage Inductor

The neutral beam injector, EF coil, OH coil and rf amplifier power supplies extract energy from a central energy storage inductor (ESI) so that the large power demands required by these systems will be buffered. The ESI design is essentially the same as that for the one described in reference 2.

The ESI design is based on an 80,000 A current limit, a peak field of 5 T, a current density of 10^7 A/m², and a stored energy of 2525 MJ, 1900 MJ of which is extractable. The ESI is a simple ring dipole (torus) with a minor radius of 0.8 m, a major radius of 4.9 m, and an inductance of 0.82 h. The design of the conductor for the ESI is based on the Rutherford cable concept

with 200 1-mm diameter wires wound on an 0.003 m by 0.097 m fiberglass support strip. The conductors would be the same as those developed for the poloidal coil design. A rectifier power supply draws an average power of 13.3 MW from the utility grid to resupply losses in the energy storage and transfer system.

VIII. ENERGY CONVERSION AND SHIELDING

The design of the energy conversion and shielding system is based, in large part, upon concepts developed in the previous EPR design.⁽²⁾ However, simplifications have been introduced to achieve some major design improvements. The principle parameters of the first wall, blanket and shield are summarized in Table 13.

A. First-Wall System

The first-wall system includes: (1) the vacuum vessel that surrounds the plasma region, (2) a liner to shield the vacuum wall from the plasma, (3) a current breaker (insulator) in the vacuum wall, (4) a plasma-aperture limiter, (5) a recirculating coolant, and (6) the vacuum wall penetrations for evacuation, auxiliary heating and diagnostics. The following design concepts are being developed to meet the requirements imposed by plasma impurity control, materials integrity, fabrication and assembly and thermal-hydraulic performance.

1. Vacuum Vessel

The first-wall design retains the self-supporting vacuum-vessel concept proposed previously.⁽²⁾ The toroidal vacuum chamber illustrated in Figure 9 is constructed of stainless steel. Sixteen three-piece segments form the 4.75-m high by 3.75-m wide noncircular cross-section torus. Each of the 48 pieces is flanged for joining to adjacent sections by a combination of bolting and welding. The vacuum wall is formed by Panelcoil* type construction with rib and spar reinforcing. Pressurized water will be used as a coolant for the first wall. Coolant inlet and outlet nozzles will be joined to headers and

* Panelcoil is a tradename of Dean Products, Inc., Brooklyn, New York.

Table 13. First Wall, Blanket and Shield

First Wall

Material	316 SS, Be-coated
Coolant	H ₂ O @ 2,200 psi
Maximum temperature	450°C
Neutron damage	11 dpa/(MW-yr/m ²)
Helium generation	215 appm/(MW-yr/m ²)
Hydrogen generation	530 appm/(MW-yr/m ²)
Nuclear heating*	13.9 W/cm ³
Surface heat flux*	0.3 MW/m ²

Blanket

Material	316 SS
Thickness	0.28 m
Coolant	H ₂ O @ 2,200 psi
Maximum temperature	500°C
Maximum coolant temperature	< 350°C
Maximum nuclear heating*	8.4 W/cm ³

Bulk Shield

Material	
inner	304 SS, B ₄ C
outer	B, C, Pb-mortar, 304 SS, Al
Thickness	
inner	0.52 m
outer	0.97 m
Maximum nuclear heating*	0.7 W/cm ³

* Based on a 14 MeV neutron wall load of 1.2 MW/m².

Figure 9. Vacuum Wall Assembly

manifolds located in the crevice volumes between the vessel and the blanket. Access to the coolant nozzles is simplified by use of the slab-type blanket and shield described in the following section.

The basic vacuum vessel, which weighs ~75,000 kg (82 tons), is supported from below on 48 columns resting on the lower shield blocks. The mass of the first-wall structure is minimized to reduce its volumetric heating and cost and to conserve space inside the TF coils. At the same time, the vacuum vessel must retain adequate strength to support the vacuum load and the weight of the first-wall structure and coolant system under the severe thermal and radiation environment of an operating reactor. The vessel wall thickness is sized to withstand local buckling, while the size and arrangement of the reinforcing ribs and spars are designed to support the vacuum and weight loads and prevent overall buckling of the vessel.

2. Current Breaker

A ceramic insulator located at a joint between two adjacent segments will serve as a current breaker in the vacuum wall. The current breaker will be formed by a coating of the insulator material on flanges of the structural segments. Yttria- or chromia-based ceramics are the prime candidates for the current breaker material. The insulator material will be applied to the substrate by the chemical-bonding process previously proposed.⁽²⁾

3. Liner

A low-Z liner will be used to shield the plasma from the high-Z elements of the structural vacuum vessel. Since no divertor is being utilized, a low-Z material facing the plasma is required to attain satisfactory plasma burns (of the order of one minute). The low-Z coating concept is being retained as the reference design; however, a separated liner is also being considered. The coating concept separates the structural requirements from the plasma contamination-minimization problems. Other advantages include minimum vacuum pumping requirements, minimal materials resource requirements and ease of fabrication, while a major concern for the coating concept relates to accidental plasma dumps on the wall. The aperture-limiter concept discussed below should tend to alleviate this problem. The beryllium coating is still optimum from a plasma impurity viewpoint, however, other low-Z materials are also

being considered, viz., BeO, B₄C, B, BN, SiC and C. Chemical sputtering of graphite and the carbides is a major concern.

4. Limiter

Plasma aperture limiters are incorporated into the present first-wall design. Sixteen poloidal limiters, one at each segment joint, are proposed. These limiters would receive most of the charged-particle transport heat load during normal operation, while the normal first wall would receive most of the radiation and neutral transport heat load. The limiters will be radiatively cooled, which will facilitate removal and replacement of the limiters, and will be partially sacrificial. The materials for the limiter must be compatible with the lower temperature first-wall coating, since material sputtered from the limiter will deposit on the wall. For this reason a graphite limiter may not be acceptable.

5. Coolant

Pressurized water is retained as the first-wall coolant, but the potential of steam is being investigated.

6. Vacuum Wall Penetrations

The major penetrations in the vacuum wall are for evacuation of the chamber and for auxiliary neutral beam heating. The same penetrations are used for both purposes. As indicated in Figure 9, sixteen ports (~ 0.1 m² each) are located on the external portion of the toroidal chamber. Twelve of these ports will be used for pumping and beam injection, while the remaining four will be used for auxiliary purposes. Other minor ports will be included as required.

B. Blanket Configuration and Considerations

The slab-type blanket shown in Figure 10 is made up of 0.28-m thick stainless steel blocks. Heat generated in the blocks is removed by pressurized water (steam is also being considered) flowing through internal coolant channels. The blanket is divided toroidally into sixteen major segments, each of which is further divided into a minimum of eleven pieces.

The blanket weighs approximately 987 Mg (1,096 tons), the heaviest piece weighing no more than 14.3 Mg (16 tons). The weight of the blanket is

BLANKET ASSEMBLY
11 PIECES

KEYNOTE

COOLANT ARRANGEMENT

Figure 10. Blanket Configuration

transmitted directly to the lower shield blocks, with the exception of the top blanket, which is suspended from the surrounding top shield blocks. Thermal expansion can be accommodated by this arrangement. Exposing the top and outer blanket blocks by removal of the respective shielding will allow for repair and replacement of the entire outer blanket in a relatively straightforward manner. Once the outer blanket is removed, the vacuum vessel may be repaired or replaced with relative ease without removing the inner or bottom shields.

Parallel coolant passages run in the poloidal direction in each of the blanket blocks. The channels are arranged to accommodate the distribution of energy deposition in the blanket. Heat is removed from each of the blanket blocks utilizing water at 2200 psi at temperatures up to 350°C. Each block is fitted with inlet and outlet coolant nozzles designed to readily connect to the header system with a minimum of manipulation.

Extensive nuclear analysis of the first wall and blanket was made in toroidal geometry. The maximum nuclear heating is 13.9 W/cm³ in the first wall and 8.4 W/cm³ in the blanket and it drops to 0.7 W/cm³ at the outer boundary of the blanket. The maximum radiation damage parameters in the first wall are 11 dpa/(MW-yr/m²) for the atomic displacement, and 215 and 530 appm/(MW-yr/m²) for the helium and hydrogen production, respectively. The helium and hydrogen production rates decrease about an order of magnitude in 13 cm, while the atomic displacement decreases only by about a factor of four in the same distance.

C. Bulk and Penetration Shielding

The bulk shield circumscribes both the inner and outer blanket regions. The inner shield consists of alternating blocks of stainless steel and B₄C as described previously. The outer bulk shield is 0.97-m thick and consists of 0.15 m of borated graphite, 0.65 m of lead mortar, 0.08 m of stainless steel and 0.09 m of aluminum, going radially outward from the blanket. The maximum heating rate in the bulk shield in regions far removed from penetrations is 0.7 W/cm³. Adequate cooling of the bulk shield is provided to limit the maximum operating temperature in the lead mortar to 80°C.

Effective shielding is provided for all penetrations that have to be accommodated in the blanket/bulk shield. The neutral beam ducts are surrounded by an 0.8-m thick local penetration shield as they emerge from the bulk shield. This penetration shield extends to the beam injectors and is tapered off in proportion with the radiation attenuation. Movable shield plugs are used where necessary to block penetrations that need not be open during the plasma burn. The concept of connecting the torus evacuation ducts to the neutral beam ducts at locations outside the blanket has several advantages over the use of separate torus evacuation ducts that extend directly to the first wall. Some of these advantages are: (1) reduction in total radiation streaming through the combined duct systems, (2) reduction of the radiation streaming through the neutral beam ducts to the interior of the beam injectors because a fraction of the neutrons and photons escape into the evacuation side channels, and (3) the fraction of the blanket volume occupied by penetrations is substantially reduced. The possibility of replacing shield plugs with local exterior shielding for the evacuation side channels is being examined.

Structurally, the design described above offers significant advantages in terms of unit fabrication costs and simplicity of support. It also provides ease of assembly and disassembly as well as access to the reactor interior for maintenance, repair and replacement of the blanket and/or vacuum vessel. A minimum of twelve major shield blocks are used to cover the blanket and vacuum vessel between the sixteen TF coils, resulting in a total of 192 pieces for the reactor shield. The total weight of the shield is estimated at 1730 Mg (1900 tons), with the heaviest single piece at less than 22.2 Mg (24.4 tons).

The geometrical arrangement, as depicted in Figure 11, shows one of the sixteen segment assemblies that encompass the blanket. The blocks are supported on three columns that extend down through the TF and poloidal coil systems to the reactor foundation. The bottom shield blocks carry the weight of the inner and outer vertical blocks to the columns. The top blocks rest on the inner and outer ones, completing the enclosure. In sequence of disassembly, the top and outer blocks are removed first, baring the reactor blanket and neutral beam penetration. From this point, those portions of the blanket that are uncovered may be removed and the vessel portions repaired or replaced without further shield disassembly.

SHIELD ASSEMBLY

Figure 11. Shield Configuration

The shield features a simple box-type construction utilizing flat surfaces only. Coolant lines will be included as needed throughout each of the shield blocks to remove the small amount of heat generated and to maintain a stable thermal environment. (No usable energy is retrieved from the shield.) Each block will have an electrically insulating coating on all surfaces to eliminate block-to-block conduction and, thereby, reduce eddy current effects.

D. Energy Conversion System

The reactor heat transport system uses pressurized water at 2200 psi (15.2 MPa) and a maximum temperature $\leq 350^{\circ}\text{C}$. The heat transport system is sized to handle 360 MW(th), which is 133% of the design value of 270 MW(th). The power plant is sized to convert 270 MW(th) into 70 MW(e), and a cooling tower handles the additional 90 MW(th) over nominal. Cooling towers are also provided to handle the 200 MW(th) discharged from the power plant and 25 MW(th) additional waste heat. The capabilities of the energy conversion system are given in Table 14.

Four reactor coolant loops, each containing a pump and steam generator, will recirculate water through a system of piping and manifolding in quantities up to $1200 \text{ m}^3/\text{s}$ (15,000 gpm). The system materials will be stainless steel

Table 14. Energy Conversion System

Heat Transport System	
Coolant	H_2O @ 2200 psi
Number of loops in reactor	4
Capacity	360 MW(th)
Power Plant Rating	70 MW(e) (270 MW(th))
Cooling Towers (4)	
Power plant	200 MW(th)
Reactor overload	90 MW(th)
Other waste heat	25 MW(th)

within the core of the reactor but may change to ferritic steel outside the region of strong magnetic fields.

Other equipment will consist of pressurizers, water clean-up system, water treatment and storage facilities. As the tritium inventory builds up in the coolant, it will be stored and disposed of as required using reasonably well-established technology. The entire energy conversion system outboard of the reactor will utilize current PWR technology in the steam generator, turbine generator, feedwater and heat rejection systems. The thermal efficiency will be approximately 25%. Preliminary calculations indicate that thermal storage requirements for the 15 s reactor down (exhaust) period can be handled by the stainless steel blanket with minor temperature fluctuations.

IX. TRITIUM

The requirements for the tritium fuel cycle system are shown in Table 15. Tritium is assumed to recycle from the wall to the plasma with a 95% efficiency, which results in an input-to-burnup ratio of 6.25 and substantially reduced quantities of tritium relative to previous estimates.⁽²⁾ If recycling does not take place, tritium input and exhaust systems must handle an order of magnitude greater quantities of tritium.

A. Fuel Cycle

The plasma exhaust, consisting of unburned deuterium and tritium plus helium, protium and varying amounts of other impurities, is collected in high speed, large capacity cryosorption pumps as described in part VI. During clearing of these pumps (presumably on a four-hour basis) the contents are rapidly transferred into a set of low residual volume cryosorption or cryo-condensation pumps. Regeneration of these pumps is accomplished by warm-up to a temperature sufficient to generate at least 10 to 20 torr of internal pressure, and the resulting gas load is removed down to 1 torr with staged bellows type compressor pumps. Alternative regeneration methods involving sealed turbomolecular pumps are also being investigated. The regenerated steam is: (1) passed over hot rare earth or transition metal beds to remove non-metallic element impurities (e.g. O, N, C), (2) passed through a cryogenic separator

Table 15. Tritium Fuel Cycle System Requirements

Fusion energy per burn pulse	20.8 GJ
Burn pulse length	65 s
Dwell time	15 s
Plant factor	50%
Input/burnup ratio	6.25
Tritium burnup	
per cycle	0.04 gm
per day	39.7 gm
per year	9.1 kg
Tritium input	
per cycle	0.23 gm
per day	246.9 gm
Tritium exhaust	
per cycle	0.19 gm
per day	207.2 gm
Tritium inventory	
storage	250 g
fuel cycle	50 g
total	300 g ^(a)

-
- a. An absolute minimum only applicable to periods of operation at low plant factors. The operating inventory at a 50% plant factor could be ten to twenty times as large.

to strip off helium, (3) transferred to a cryogenic distillation cascade for enrichment, and (4) thereafter passed on to storage or fueling systems. The fuel cycle contains numerous instrumented check points where fuel composition integrity is verified prior to transfer to subsequent processing stages.

The distillation cascade produces three outlet streams; a waste stream consisting mostly of deuterium and protium, a stream for the neutral beams consisting of 99+% deuterium, and a mixed DT stream enriched in tritium. The waste stream is processed for packaged disposal. The deuterium stream is passed to the neutral beam injector system. The DT fuel stream is sent to the storage units, which are basically getter-type beds. The beds then release DT fuel to a fuel blender, which adjusts the isotopic ratio for fuel delivery. The fuel is delivered to the plasma by injection of frozen DT pellets or by gas injection.

B. Tritium Containment Systems

Tritium systems are designed with the philosophy of multiple containment in order to ensure that unacceptable amounts of tritium are not released to the environment. The primary level of containment consists of the higher quality fuel cycle hardware itself. A secondary level of containment is provided by enclosing all tritium handling systems in gloveboxes or double-walled pipes. The jacketed enclosures are slowly purged with inert gas and the gas is scrubbed to remove tritium. The integrity of primary containment is continuously checked by monitoring the purge streams. Should a release of tritium out of secondary containment occur, a 10^4 cfm emergency cleanup system is provided such that the reactor building atmosphere and all ancillary atmospheres can be processed until the tritium levels are $< 5.0 \mu\text{Ci}/\text{m}^3$.

The water coolant will be monitored for tritium content. Significant amounts of tritium could diffuse into the first-wall coolant water. It is estimated that at least 20 Ci/day of tritium passes into the first-wall coolant. This corresponds to an accumulation rate of 0.2 m Ci/l-day or a total accumulation of about 4 Ci/l in ten years of operation, which is less than is commonly encountered in heavy water fission reactors.

X. MATERIALS LIMITATIONS

It appears at this time that there are no materials limitations that preclude the viability of the EPR design. However, there are five major areas where more data are needed before a final design is committed.

The first of these involves the capability of austenitic stainless steel to operate successfully over the temperature regime from 300 to 500°C. The form of radiation damage changes dramatically over this temperature range, going from a high concentration of very small point defect clusters at 300°C to an array of relatively large faulted dislocation loops and voids at 500°C. Assuming an allowable swelling of 10% and a reduction in ductility to 0.5% uniform strain, the region of the wall at temperatures in excess of 400°C would be limited to approximately 4 MW-yr/m² of service. Thus, at the nominal* EPR wall loading of 0.64 MW/m² and plant factor of 50%, the first wall would require replacement every 12.5 years. Defining the lifetime of the regions at temperatures near 300°C is more difficult, as considerably less information is available. The two properties that need better definition at this irradiation temperature are ductility and fatigue. The design limit of 0.5% uniform ductility is based upon fission reactor data. There are several factors that could cause a higher limit to ultimately be set for EPR. A more conservative limit of 1% ductility would correspond to 2.5 MW-yr/m², which would require wall replacement every 7.8 years. Fatigue tests, in particular, at a variety of conditions will be needed prior to a final definition of a component replacement schedule.

The second area deals with the fact that the EPR blanket and shield regions will be heavily welded structures. More needs to be known on how irradiation affects the fatigue and tensile properties of welded austenitic steels.

The third area concerns the radiation damage of high temperature inorganic insulators, since the EPR design includes a ceramic circuit breaker in the first wall. While yttria-based materials look very attractive for this application at this time, much more needs to be done to develop the necessary property data before an accurate estimate of an expected lifetime can be established. Existing information does indicate that the yttria is stable to at least 2 MW-yr/m², or 6.25 years of EPR service.

* The maximum wall load is 1.2 MW/m², but the average wall load over the burn pulse is 0.64 MW/m².

The fourth area where improved information would help is in the selection of a cryogenic insulator for use in the superconducting magnets. Commonly used organic insulators may be limited to $\sim 10^9$ rads of exposure, which dictates the use of a thicker than otherwise optimum blanket and shield. More detailed information is needed on conventional glass-bonded epoxy insulators as well as on attractive alternates such as glass-bonded mica.

The fifth area concerns the development of a low-Z first-wall coating that either is long-lived or can be easily replaced. Sputtering rates, especially at lower projectile energies, for both deuterium and helium are needed before the burn cycle can be fully defined.

XI. REACTOR ENGINEERING

Four major engineering aspects of the EPR are described summarily: the reactor support and torque reactions, a secondary vacuum enclosure, the assembly of the reactor and the remote maintenance and repair systems.

A. Structural Support Systems

Two areas of reactor structural support have been reviewed - support of the blanket, shield and vacuum vessel and support and restraint of the various magnet coils. Support of the reactor shield and its contained blanket and vacuum vessel has been simplified over that of the previous EPR design.⁽²⁾ Utilization of the slab-type geometry for the blanket and shield allows the bottom shield blocks to be used as the base structure, supported by three to four simple stationary columns per segment of reactor (sixteen segments total). Inner and outer vertical blanket and shield form column walls, which support the top section as a lintel. The vacuum vessel is supported through thin columns to the lower portion of the bottom shield and may be adjusted or replaced easily from beneath the reactor. All aspects of service, repair and replacement of components are simplified by this arrangement.

Having reduced the number of major reactor ports from 44 to 16 and with the simplification of the blanket and shield systems, the choice of the torque shell concept for restraining the TF coils has been selected in favor of the

torque frame system.* Sixteen shear webs provide continuous support for the TF coils. An overhead tree-type structure supports and restrains the upper EF and OH poloidal coils. Side columns support the outer poloidal coils, while a lower network of removable structures provide for the remaining coils. The lower coils are held captive by the TF coil system and can be repaired (re-wound if necessary) in a lowered position, while all others may be removed to a remote area.

B. Secondary Vacuum System

A second independent vacuum enclosure is contemplated for the reactor. This enclosure would utilize the torque shell structure to completely enclose the blanket/shield system and cryosorption pumps within the confines of the TF coils. A secondary enclosure would encompass each neutral beam injector system. The structures and panels required would be simple and relatively inexpensive. Advantages of this concept are many-fold, but the principle ones are safety and operability.

Safety features include secondary containment of tritium, with the ability to monitor an intermediate zone. Cleanup of tritium leakage would be quick and safe and require less equipment. In terms of reactor maintenance, this would obviate the need for personnel breathing apparatus. The major operational advantage is the capability of continued operation of the reactor with minor leaks, as allowed by the monitoring feature.

C. Reactor Assembly

Assembly of the reactor is a critical portion of the overall design and is factored into the design at an early stage. Features that have been considered are:

Reactor Mock-up for testing of preassembly equipment, such as that used for TF coil cryostat erection and final fabrication and for development of procedures for their installation and repair.

* See Appendix D of reference 2 for a discussion of these two options.

Coil Winding facilities for the TF coils on site, with rigging and access into the reactor containment. Space and lifting equipment to fabricate the larger diameter OH and EF coils, 20-m OD (66 feet), within the reactor containment shell. Facilities and space to completely rewind the lower-inner EF and OH coils held captive by the TF coil network.

In addition to the various integrated design features, a very detailed assembly schedule is required, not only to assure proper assembly of the reactor but to assure that, once radioactive, all procedures for remote or semi-remote repair of every component will be available. Further, in order to assemble the reactor as quickly as possible, a master schedule is required to utilize limited space and equipment (such as the overhead cranes) as efficiently as possible. The reactor assembly plan in reference 2 has been adapted to the new design.

D. Remote Maintenance

The basic approach to remote maintenance is that all maintenance operations are to be performed either by semi-remote or fully remote means. All maintenance and repair operations in and near the reactor at shutdown must be accomplished with remote apparatus, due to the residual radioactivity. If the neutral beam injectors are adequately shielded, this remote maintenance could be performed from inside the reactor building. The apparatus used may be anything from a 1 to 2-m extension for an impact wrench ("hands on" maintenance) to remotely guided robots that transverse the reactor vessel interior.

Various support facilities have been identified for remote maintenance activities, among which are: a one-fifth segment mock-up of the reactor; two hot cells, one for neutral beam repair with atmospheric control and a second for blanket/shield block or vacuum vessel repair and use for irradiated experiments; a neutral beam test area; a special shielded cab fitted with manipulators on the polar crane; basement facilities for handling components beneath the reactor; and various coil winding facilities.

Special design considerations have been given to personnel safety in doing maintenance work. In addition to shielding, a secondary vacuum vessel has been placed over the reactor and the neutral beam injectors have been

doubly encased. This feature will allow personnel to enter the reactor room after shutdown without the need for special breathing apparatus.

Repair and replacement of the blanket and vacuum vessel will be simpler, and less time consuming than in the previous EPR design⁽²⁾ due mainly to the smaller number of shield and blanket blocks required in the slab-type shield geometry. Internal machines for repair of the first wall are considered optional at this time. However, inspection robots will be used as a prime diagnostic tool.

XII. REACTOR FACILITY

The overall reactor facility is shown in Figures 12 and 13. It consists of the reactor building and seven major support buildings and structures. The major reactor support facilities are located in an attached building adjacent to the main reactor cargo doors. This building houses the repair/maintenance hot cell, the mockup/test equipment and other special facilities for initial winding and cryostat preparation of the TF coils. Adequate cranes and access to equipment are provided for handling all reactor components prior to their installation. A third large structure also located adjacent to the reactor building houses the reactor magnet power supplies and compressors for the cryogenic units. Power leads are run through seals in the reactor containment to the various reactor components. An energy storage ring, remotely located, is also part of the reactor power distribution system. The facility also has a tritium facility for storage, preparation and reprocessing of reactor fuel.

A 70-MW(e) power plant is a major facility feature. Reactor heat up to a maximum of 360 MW(th) is handled by a pressurized water heat transport system. Other facility structures include a facility air handling building, an office building housing the control center, a waste heat cooling tower array and a 70-MW(e) power substation.

Figure 12. ANL Experimental Power Reactor Facility Elevation

Figure 13. Plan View of Facility

A. Reactor Containment Building

The reactor building is designed to house and structurally support the reactor and its attached ancillary equipment. In addition, it provides: space and equipment for initial reactor assembly and subsequent remote maintenance and repair, an area and equipment for neutral beam testing, biological shielding, secondary atmospheric containment and experimental equipment space. A circular building, 57-m in diameter by 51-m in height, having an elliptical-shaped roof and flat base constructed of reinforced concrete, was selected. The wall, roof, floor and base of the building are 1.52-m thick reinforced concrete, which provides the structural support as well as the biological shielding required. A 175-ton capacity polar crane services the entire reactor floor. This crane is designed for initial reactor assembly as well as subsequent operations, with a capacity to maneuver the heaviest reactor component, three assembled TF coils.

A 1-cm thick steel membrane liner provides a continuous vapor barrier primarily for secondary tritium containment. Two cargo-sized sealed doors and two personnel doors (one air lock) are used for access. The reactor support consists of a central pedestal surrounded by an annular ring abounded by the reactor service floor. Both pedestal and ring structures extend downward 10-m to a sub-building foundation pad. A sub-reactor basement area houses: the TF coil cryogenic cold box supply, the TF coil energy dump resistors, neutral beam high voltage power source, vacuum pumps and blowers, first wall and shield primary coolant transport and coolant storage system, magnet power supply cables and space for diagnostic and experimental facilities. The building is designed to withstand a vacuum to less than 0.2 atmospheres.

B. Mockup Maintenance Building

This building is approximately 60-m (197 feet) long, 34-m (110 feet) wide and 18.3-m (60 feet) tall. It is covered by a traversing crane whose capacity is 100 tons. Three reactor services are performed in this building: first as an initial assembly area for items such as winding TF coils, cryostat assembly, shield and blanket block inspection, torus preassembly, etc. A second permanent function is for remote assembly, testing and repair support.

A large hot cell is provided for neutral beam maintenance. Equipment in the hot cell includes a crane mounted manipulator and hoist, a rail system for maneuvering neutral beam injectors, a penetration access directly into the neutral beam test area, a normal contingent of conventional manipulators and viewing windows, roof access to the building crane hoist and a basement storage area access. There is a second smaller hot cell for repair and modification of activated reactor components. This hot cell, in addition to serving the reactor needs, can accommodate experimental equipment. A laboratory/office wing is provided at one end of the building for maintenance of equipment, experimental areas and storage. A machine shop will be included.

C. Power Supply Buildings

A large building measuring 30-m (100 feet) by 80-m (260 feet) by 12.2-m (40 feet) tall houses the power supplies for the magnets and auxiliary heating systems and the compressors for the cryogenic systems. The building is adjacent to the reactor building and equipment is arranged to minimize the power lead runs to strategic components such as the neutral beam injectors. Two overhead cranes traverse the entire building on individual bridges. A partial basement adjoining the reactor building is also included. An energy storage ring maintained at liquid He temperatures is part of the power supply system. It is 20-m in diameter and located about 80-m away from the facility, slightly below grade.

D. Tritium Facility

The tritium facility consists of a steel containment building 23-m (75 feet) in diameter by 37-m (121 feet) high and a two-level attached laboratory area of approximately 186 m^2 (2000 ft^2). The containment building is used for fuel storage and reprocessing equipment. Two air locks allow personnel entry while maintaining the seal integrity. This facility will also house equipment for handling inadvertent release of tritium in the reactor containment building.

E. Power Plant

A 70-MW(e) power plant is provided adjoining one quadrant of the reactor building. It consists of an 18-m wide by 14-m long by 18-m high steam generator structure with appropriate tritium containment features. Steam generated here

is piped into the adjoining turbine hall and heater bay, 30-m wide by 60-m long by 30-m high, which houses the turbine-generator set, condenser and feedwater system and resembles a PWR power plant. The plant is shown in Figure 13 with a pump station, a row of forced draft cooling towers, control room and power substation capable of handling 70 MW(e).

F. Office/Control Building

A 32-m (100 foot) by 48-m (135 foot) building serves as a central control station, office and laboratory space. The building is four stories high having a 9.2-m (30 foot) by 18.3-m (60 foot) control room two stories tall.

G. Air Handling Facility

An air handling facility is located adjacent to and integrated with the tritium facility. This facility will handle the reactor building air circulating system. It has a lower air intake plenum and tall exhaust stack. Internally it has isolation equipment for emergency use for processing of tritiated air from the reactor containment building.

H. Cooling Towers

Four or more forced draft cooling towers are provided to dissipate 200 MW(th) of low-grade heat discharged from the power plant under normal operating conditions, 90 MW(th) of high-grade heat when the reactor is operating at 133% of nominal power and 25 MW(th) of low-grade waste heat. This tower system will house secondary coolant pumps and a complete purification system. One tower serves the reactor with a relatively high ΔT spread. The other three units run closer to ambient conditions, supplying water to cool power supplies, lead cables and other equipment.

I. Power Substation

A 70-MW(e) substation with 132 kV incoming power is provided. This is adequate in the advent that the reactor is operated without the power plant.

XIII. COST ESTIMATE AND SCHEDULE

A. Cost Estimate

A preliminary cost estimate is shown in Table 16. Significant cost reductions have been achieved, relative to the previous EPR design.⁽²⁾ These cost reductions are, for the most part, attributed to design improvements and innovations, reduction in the size of the reactor and reduction in the power handled during reactor operation. Major cost reductions attributed to the reactor are as follows:

	<u>Current Design</u>	<u>Previous EPR Design</u> ⁽²⁾
Overall reactor size	20-m dia. x 14-m high	25-m dia. x 19-m high
Blanket	143 m ³ volume in 176 simple slab-type blocks	173 m ³ volume in 272 contoured blocks
Outer shield	768 tons/240 plane surface blocks	1130 tons/400 curved surface blocks
TF coils	10 m x 6.8 m bore/ ~ 46 tons	12.6 m x 7.8 m bore/ ~ 208 tons
Vacuum system	24 units, single pump, using beam ducts	32 units, dual pump, separate ducts

Additional cost reductions are attributed to changes in the reactor support facilities and power requirements.

- Reactor Containment Building - The smaller sized reactor requires a smaller sized containment building of approximately one-half the total volume with much smaller cranes and other support equipment.
- Power - The power plant was reduced from 150 to 70 MW(e); the capacity of the heat transport and reject systems were reduced; and the ratings of the energy storage and transfer systems for the plasma heating and pulsed magnet systems were reduced.
- Tritium Facilities - Use of a secondary vacuum vessel reduces the reactor cleanup system requirements substantially.

Table 16. Plant Capital Investment Direct Cost Estimate
(in millions of 1977 dollars)

<u>1.0 Structures & Site Facilities</u>		
1.1	Site improvements	\$ 2.2
1.2	Reactor building (mock and hot cell)	30.8
1.3	Electric power conversion building	3.1
1.4	Cryogenic building	1.2
1.5	Energy storage ring building	0.8
1.6	Tritium separation building	3.1
1.7	Heat exchanger building	1.3
1.8	Turbine generator building	2.2
1.9	Office and control center	<u>3.5</u>
	Subtotal	\$ 48.2
<u>2.0 Reactor</u>		
2.1	Magnet systems	\$ 75.0
2.2	First-wall system	8.2
2.3	Blanket system	9.9
2.4	Shield system	23.6
2.5	Support frame and secondary vacuum vessel	5.3
2.6	Vacuum systems	28.0
2.7	Neutral beam injectors	22.0
2.8	rf heating	--
2.9	Instrumentation	<u>12.0</u>
	Subtotal	\$184.0
<u>3.0 Reactor Plant Facilities</u>		
3.1	Tritium facilities	\$ 30.0
3.2	Electric power supplies	70.0
3.3	Heat transport and exchange system	9.5
3.4	Instrumentation and control	10.0
3.5	Remote maintenance equipment	18.0
3.6	Cryogenic refrigeration supply	18.0
3.7	Storage ring	<u>10.0</u>
	Subtotal	\$165.5
<u>4.0 Turbine Plant</u>		
4.1	Turbine generator - 70 MW(e)	\$ 9.8
4.2	Heat rejection systems	3.2
4.3	Auxiliary power supply - 5 MW	3.2
4.4	Site power distribution	1.8
4.5	Site power feedlines	<u>1.0</u>
	Subtotal	\$ 19.0
	Total	\$416.7
	Engineering (19%)	79.0
	Contingency (25%)	<u>104.7</u>
	Grand Total	\$599.4

B. Schedule

The construction schedule shown for the earlier EPR design⁽²⁾ is approximately five years from beginning of site construction to initial startup. The reactor assembly period is approximately 32 months. Utilizing some of the new design features, the assembly period of this upgraded EPR may be shortened as much as five months, and the total construction period may be shortened as much as 9 months. The potential reduction in reactor assembly schedule can be attributed to the following:

- Removable Vessel - The vessel may be preassembled, checked and tested prior to erection;
- Blanket and Shield - Utilization of the slab-type structures allows a simplified support system and one which can be quickly installed (400 versus 700 pieces);
- Reactor Penetrations - Fewer penetrations through the reactor shield are required - 12 versus 44 for neutral beam injection and vacuum pumping. The alignment and special fittings required for the vacuum ports will be reduced significantly. Those that remain may be pre-assembled independently of the reactor;
- Neutral Beam Injection - Using radially aimed neutral beam ports simplifies the assembly of the shield and blanket around these ports;
- TF Coils - With the significant reduction in coil weight - 46 versus 208 tons each - TF coils may now be assembled in groups of three within the reactor building.

Basically, more assembly work can be done concurrently and more large components can be preassembled and checked during construction, relative to the previous EPR design.

The potential reduction in facility construction time comes as a result of not allowing as much lead time for the pre-reactor mockup assembly demonstration as in the previous design.⁽²⁾ Building construction speed now becomes limiting. A simplified schedule is indicated in Figure 14.

Figure 14. Reactor Construction Schedule

XIV. TEST FACILITY POTENTIAL.

The EPR could serve as an extremely effective materials test facility for irradiating fusion reactor materials. It has the distinct advantage that a large test volume and prototypic irradiation conditions for all materials of interest are inherent to its operation. As shown in Table 17, at a 1 MW/m² wall loading level and a 50% plant factor, property information over a 6 to 60 dpa damage range will be generated over a 1 to 10 year period. Thus, statistically significant quantities of mechanical property data on all materials of interest will begin to become available in 1991, assuming an EPR operational date of 1990. The large test volume available will allow many in-situ mechanical property tests to begin immediately so that an appropriate fatigue and crack propagation data base can be rapidly established.

To put in perspective the ultimate needs of the fusion program for test volume, it should be recognized that the LMFBR alloy development program alone has planned to use $\sim 10^5$ cm³ of test space. The fusion program may well need more since it must more fully establish in-reactor mechanical properties on a wider range of materials (e.g. insulators, coatings, structural, limiter, etc.) than have concerned the LMFBR program. Assuming that it is required to irradiate 10^5 cm³ of material to 60 dpa, which is a dose representative of high

Table 17. A Comparison of Materials Irradiation Test Facilities

Facility	$\frac{\text{dpa}}{\text{sec}} \times 10^{-7}$ SS	Duty Factor %	$\frac{\text{dpa}}{\text{yr}}$ SS V	appm He/yr SS V	Test Vol. cc	Irrad. Capacity $\frac{\text{dpa} \cdot \text{cm}^3}{\text{yr}}$	Irradiation Capacity $\frac{\text{appm He} \cdot \text{cm}^3}{\text{yr}}$ SS V	Time ^b required (years) to irradiate 10^5 cm^3 to 60 dpa SS V	Date Available
1 $\frac{\text{MW}}{\text{m}^2}$ EPR	3.6	50	5.6 6.7	108 30	$2.7^a \times 10^6$	1.5×10^7	2.9×10^8 8.1×10^7	10^c 10^c	1990
2 $\frac{\text{MW}}{\text{m}^2}$	7.2	50	11.2 13.4	216 60	2.7×10^6	3.0×10^7	5.8×10^8 1.6×10^8	5^c 5^c	1990
ORR (21)	4.0	88	11.0 14.2	120 6	1.8×10^4	2.0×10^5	2.2×10^6 1.1×10^5	25^d Not Applicable	1977
EBR-II (22) (core center)	14.0	71	31.0 37.6	3.3 0.2	$\sim 600/\text{ass'y}$	$\frac{1.9 \times 10^4}{\text{ass'y}}$	$\frac{2.0 \times 10^3}{\text{ass'y}}$ $\frac{120}{\text{ass'y}}$	Not directly useful, gas generation rate too low	1977
RTNS-II (23) (2×10^{13} n/cm ² ·sec)	0.4	90	1.2 1.0	45 18	1	1.2	45 18	$5^d \times 10^6$ 5×10^{6d}	1978
INS (23)	2.1	80	5.2 4.3	195 78	3	15.6	585 234	$3.6^d \times 10^5$ 3.6×10^{5d}	1981
(D,L1) (23) 10^{15}	20.6	80	5.2 NA	800 NA	10	520	8000 NA	$12,000^d$ NA	1983
10^{14}	2.1	80	5.2 NA	80 NA	300	1560	2.4×10^3 NA	3.6×10^{4d} NA	
5×10^{13}	1.0	80	2.6 NA	40 NA	1000	2600	4.0×10^4 NA	$2,000^d$ NA	

a. Based on use of 10-cm region adjacent to 3/4 of wall.

b. Volume estimate for FBR alloy development program plan, dpa estimate for minimum 10 MW-yr/m² operation.

c. Limited by time to achieve goal dpa.

d. Limited by time to irradiate needed volume of specimens.

damage conditions, it is illustrated in Table 17 that only ORR of the other possible irradiation sources can make any significant contribution to the gathering of engineering data. However, even this valuable facility is limited in test volume and in the fact that it is restricted to being able to simulate irradiation conditions for only nickel bearing steels.

This may be an important restriction since recent systems studies have shown the economic potential of vanadium alloys. A fusion-based test reactor is essential to their full development as it is to the development of molybdenum, titanium and niobium alloys. In addition to these structural materials, a fusion environment is required for the full development and proof testing of first wall insulators and low-Z coatings.

The nominal EPR design described in this report has an average 14 MeV neutron wall load over the burn pulse of 0.64 MW/m^2 . However, there are a number of possible ways to increase the wall load, e.g. tailoring the burn pulse with beam heating, operating at higher β_t or higher B^{TFC} (see Tables 3 and 4). Thus, the EPR would seem to have the potential of satisfying the materials irradiation requirements for the DMFE program.

The EPR would, of course, serve the function of an engineering test facility. Provision is made in the design to insert one or more blanket test modules that would be used to demonstrate tritium breeding, advanced materials and coolant concepts, etc. Four large ($\sim 0.7 \text{ m}^2$) ports are provided for large-scale experimentation (e.g. rf heating).

XV. DESIGN SUMMARY

Relative to the previous design, ⁽²⁾ a simpler, more compact and less expensive design has been identified for the EPR. The compactness was achieved, without degrading the performance, by predicating the design upon an elongated plasma ($b/a = 1.3 - 1.7$), which may operate with $\beta_t \approx 8-10\%$. Notable simplifications were achieved by adopting a slab-type blanket and shield configuration and by pumping the toroidal vacuum chamber through the neutral beam ports.

The cost reduction was a result of reductions in size and systems requirements. The principle design parameters for the EPR are compared with the previous EPR design⁽²⁾ in Table 18. The reactor design is depicted in Figures 15 and 16.

Table 18. Comparison of EPR-77 and EPR-76⁽²⁾ Parameters

	<u>EPR-77</u>	<u>EPR-76</u>
<u>Geometry</u>		
Major radius, R (m)	5.00 (4.70*)	6.25
Plasma minor radius, a (m)	1.67 (1.34*)	2.10
Plasma elongation, b/a	1.3 (1.65*)	1.0
Aspect ratio, A	3.0 (3.5*)	3.0
Vacuum chamber volume, V_c (m ³)	450	711
Blanket thickness, Δ_B (m)	0.28	0.28
Shield thickness		
inside, Δ_S^i (m)	0.52	0.58
outside, Δ_S^o (m)	0.97	0.97
TF coil bore (m)	6.8 x 10.0	7.8 x 12.6
<u>Steady-State Plasma</u>		
Average toroidal beta, β_t	0.08	0.048
Safety factor, $q(\psi_{eff})$	3.0	3.0
Plasma current, I (MA)	7.3	7.6
Average ion density, $\bar{n}$ (10 ²⁰ /m ³)	1.3	0.9
Average electron temperature (keV)	8.0	10.0
Effective charge, Z_{eff}	1.7	1.7
Energy confinement, τ_E (10 ²⁰ s/m ³)	3.4	2.4
Fusion power, P(th) (MW)	620	638
14 MeV neutron wall load, P_W (MW/m ²)	1.2	0.86
<u>Power Performance</u>		
Nominal burn pulse length, Δt_{burn} (s)	65	55
Dwell time, Δt_{down} (s)	15	15
Fusion energy per burn pulse, U_{th} , (GJ)	20.8	16.0
Thermal-to-electrical conversion efficiency, η_{th} , (%)	25	30
Gross electrical power, P (MW(e))	67	72
Net electrical power, P_{net} (MW(e))	39	30

* Alternative design option.

Table 18. Comparison of EPR-77 and EPR-76⁽²⁾ Parameters (Continued)

	<u>EPR-77</u>	<u>EPR-76</u>
<u>Plasma Driving and Heating System Requirements</u>		
Volt-second to plasma		
OHC (V•s)	55	85
EFC (V•s)	31	50
Energy transferred (start-up)		
OHC (MJ)	430	1200
EFC (MJ)	732	1500
Maximum field in OHC, B_{OH} (T)	4.0	5.0
Maximum field rise in OHC, $\dot{B}_{OH}$ (T/s)	4.5	6.7
OH current reversal time, Δt_{OH} (s)	2.0	2.0
Neutral beam (rf) power, P_B (MW)	40	60
Neutral beam (rf) heating time, Δt (s)	6.5	4.8
<u>Neutral Beam Injection System</u>		
Beam composition (D^+ , D_2^+ , D_3^+)	0.75, 0.18, 0.07	0.75, 0.18, 0.07
Ion beam power, P_{beam} (MW)	261	441
Deuteron energy, U_B (keV)	180	180
Number of injectors	12	12
Gas load/injector (Torr-l/s)	61	110
Cryo-panel area/injector (m ²)	80	100
Zr-Al panel area/injector (m ²)	40	-
Power efficiency (%)	37	34
Net power from energy storage (MW)	119	207
<u>Energy Storage and Transfer</u>		
Energy transfer (MJ)		
OH homopolar bank	534	1200
OH IC bridge	350	600
EF IC bridge	732	1500
NB SCR	770	1000

Table 18. Comparison of EPR-77 and EPR-76⁽²⁾ Parameters (Continued)

	<u>EPR-77</u>	<u>EPR-76</u>
<u>Energy Storage and Transfer</u> (Continued)		
Maximum power (MW)		
OH homopolar bank	840	1900
OH IC bridge	56	66
EF IC bridge	226	416
NB SCR	120	207
S/C energy storage inductor		
Energy stored (GJ)	2.5	3.2
Energy transferred (GJ)	1.9	2.4
Makeup power (MW)	13	21
Power substation capacity	70	150
<u>Toroidal-Field Coils</u>		
Number	16	16
SC/stabilizer/support	NbTi/Cu/SS	NbTi/Cu/SS
Design field at conductor, B^{TFC} (T)	9	10
Design temperature (K)	4.2	3.0
Design strain, ϵ (%)	0.2	0.09
Stability	cryostatic	cryostatic
Cooling	pool boiling	pool boiling
Ampere-turns (MAT)	96	134
Average current density over Cu (A/cm^2)	5090	3660
Stored energy (GJ)	11.4	30.0
Weight per coil (ton)	46	208
<u>Poloidal-Field Coils</u>		
SC/stabilizer for OH and EF coils	NbTi/Cu	NbTi/Cu
Ampere-turns (MAT)		
OH coils	38	67
EF coils	18	19
Average current density (A/cm^2)		
OH coils	2110	2640
EF coils	2640	2946

Table 18. Comparison of EPR-77 and EPR-76⁽²⁾ Parameters (Continued)

	<u>EPR-77</u>	<u>EPR-76</u>
<u>Poloidal-Field Coils</u> (Continued)		
Maximum current (kA)		
OH coils	64	80
EF coils	80	80
Power supply voltage (kV)		
OH coils	30	48
EF coils	14	21
Magnetic energy OH + EF + Plasma (MJ)	1384	2262
Cu Initiation-Trimming Coils		
Breakdown voltage, V_{LOOP} (V)	450	500
Voltage pulse length, Δt_{BD} (ms)	10	6
<u>Toroidal Vacuum System</u>		
Evacuation ducts	beam ducts	separate ducts
Number of cryosorption pumps	12 (24)	32 (64)
Dwell time, Δt_{down} (s)	15	15
Gas load (10^3 Torr-l)	3.7	2.6
Rated speed/pump (10^4 l/s)	8.0	2.5
Pumping speed (10^5 l/s)	1.9	2.5
<u>Blanket and Shield</u>		
<u>Blanket</u>		
Material	316 SS	316 SS
Thickness (m)	0.28	0.28
Coolant	H ₂ O @ 2200 psi	H ₂ O @ 2200 psi
Maximum structural temp. (°C)	500	500
Maximum heat deposition (W/cm ³)	8.4	3.5
Weight (tons)	1096	1386

Table 18. Comparison of EPR-77 and EPR-76⁽²⁾ Parameters (Continued)

	<u>EPR-77</u>	<u>EPR-76</u>
<u>Blanket and Shield (Continued)</u>		
<u>Shield</u>		
Material		
Inner	304 SS/B ₄ C	304 SS/B ₄ C
Outer	SS/Pb-mortar/C/Al	SS/Pb-mortar/C/Al
Thickness (m)		
Inner	0.52	0.58
Outer	0.97	0.97
Weight (tons)	1900	2970
<u>Radiation Damage to First Wall</u>		
Design lifetime		
Integrated 14 MeV neutron load (MW-yr/m ²)	4.0	2.5
Years*	12.5	10.0
Neutron fluence (10 ²² n/cm ²)	9.6	6.0
Neutron damage (dpa)	44	28
Helium generation (appm)	860	538
Hydrogen generation (appm)	2120	1325
<u>Tritium Fuel Cycle</u>		
Burnup per day (gm)	40	64
Input per day (gm)	247	3600
Exhaust per day (gm)	207	3536
Annual tritium consumption (kg)	9.1	16.0
Tritium inventory (kg)	0.3	1.5
<u>Energy Conversion</u>		
Heat transport system capacity (MW(th))	360	600
Power plant rating (MW(e))	70	100
Cooling tower capacity (MW(th))	315	600

* Based upon an average 14 MeV neutron wall load over the burn pulse of 0.64 MW/m² and 0.50 MW/m² for EPR-77 and EPR-76, respectively, and a plant factor of 50% in both cases.

Table 18. Comparison of EPR-77 and EPR-76⁽²⁾ Parameters (Continued)

	<u>EPR-77</u>	<u>EPR-76</u>
<u>Costs</u>		
Structures and site facilities	\$ 48.2 M	\$ 67.3 M
Reactor	184.0	248.6
Reactor plant facilities	165.5	245.0
Turbine plant and auxiliaries	<u>19.0</u>	<u>18.0</u>
Total	\$416.7 M	\$578.9 M
Engineering	79.0	84.0
Contingency (25%)	<u>104.7</u>	<u>144.7</u>
Grand Total	\$599.4 M	\$807.6 M

Figure 15. Elevation View of Reactor

Figure 16. Experimental Power Reactor Plan View

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