OAK RIDGE EPR REFERENCE DESIGN

M. Roberts
for the Oak Ridge EPR Project Team
Oak Ridge National Laboratory*
Oak Ridge, Tennessee 37830

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Abstract

A Tokamak EPR reference design is presented as a basis for further study leading to a conceptual design. The set of basic plasma parameters selected, namely, minor radius of 2.25 m, major radius of 6.75 m, magnetic field on axis of 4.8 T and plasma current of 7.2 MA, should produce a reactor-grade plasma with a significant neutron flux even with the large uncertainty in plasma physics scaling from present experience to large sizes. Neutronics and heat transfer calculations coupled with mechanical design and materials considerations were used to develop a blanket and shield capable of operating at high temperature, protecting the surrounding coils, being maintained remotely and, in a few experimental modules, breeding tritium. The toroidal field coil design developed for a maximum field of 11 T at the winding combines the use of multifilamentary Nb_3Sn and NbTi superconducting cables (in high and low field regions, respectively) with forced flow of supercritical helium enclosed in a steel conduit. The poloidal magnetics system is specially designed both to reduce the toal volt-second energy requirements and to reduce the rate of field change at the toroidal field coils. The reference design was synthesized from the information developed and evaluated in our prior scoping study.[1]

Plasma

There are many ways to arrive at a set of reference parameters for a future tokamak system. The same difficulty is encountered in most of the procedures, namely, that a proven relationship between attainable plasma parameters and system size does not exist at present for the type of device under consideration. Results have been obtained from present day devices coupled with theoretical predictions, but the connection between these results and the operating parameters to be attained in an EPR-size system involves much uncertainty.

The theoretical basis used here to arrive at a reference design is straightforward and it provides a machine size which should produce reactor-grade plasma in terms of presently accepted,[2] reactor modeling calculations. The steps taken in determining the reference design are described below.

We can define safety factor q and central axis magnetic field strength B_T, as the following,

$$q = (1/A)\ B_T/B_p \quad ,$$

where A is the aspect ratio and B_p is the poloidal magnetic field strength.

$$B_T = B_{max}\ (1 - 1/A - \Delta/R_o) \quad ,$$

where B_{max} is the maximum magnetic field strength at the coil surface, R_o is the major radius and Δ is the distance from plasma edge to coil inner bore. Using these relations, we find plasma current

$$I \propto a\ [(A - 1) - \Delta/a] \quad ,$$

where a is the plasma radius. In reactors $\Delta \ll a$ is possible and the effect of Δ is, therefore, of little importance. In the EPR, however, $\Delta \sim 1.55$ m which suggests a minimum $a \sim 2.0$ m so that efficient use can be made of the volume within the toroidal coils.

Table 1 displays a summary of the key plasma related parameters in the ORNL EPR. The basic TF coil size is determined from the a and Δ presented above. The number of coils is minimized for access purposes while providing an acceptable ripple as considered in greater detail elsewhere.[3]

Table 1. ORNL EPR Reference Design-Plasma Related Parameters

Plasma radius, a (m)	2.25
Major radius, R_o (m)	6.75
Safety factor at r = a, q	2.5
Plasma current, I (MA)	7.2
Plasma shape	Circular
On axis toroidal field, B_T (T)	4.8
TF ripple, at θ= 0; r = a, δ (%)	2.2 ($\sim r^4/a^4$)
OH/EF configuration	*
OH flux change, $\Delta\phi$ (Wb)	185
Injection power capacity, P_b (MW)	100
Deuteron energy, E_b (keV)	200
Output power, driven, P_d (MW,th)	400
Output power, ignited, P_i (MW,th)	200**
Duty cycle (%)	>50
TF coil opening, horiz. x vert. (m)	7.4 x 10.2
Number of TF coils, N_c	20
TF coil shape	Non-circular

*Air core with electromagnetic shielding of TF coils
**For $\beta_p \sim A^{1/2}$

The reference system will provide containment of both 200-keV deuterons and 3.5 MeV alpha particles that is adequate from the plasma heating point of view. Additional analyses are required to determine the finite gyro-radius effects to be expected and the plasma response to them.

Figure 1 shows the P_α/P_{loss} ratio vs. time for 3 toroidal field values, 100 MW of injection, and for trapped-particle losses reduced by a factor of 10. In the reduced loss cases, ignition occurs at both 4.8 T and 4.4 T. Plasma conditions at both field levels are very similar at ignition. The 3.7 T case does not ignite and the beam power is reduced after 2 seconds to prevent β_{pe} from exceeding 1. The benefit of increased B_T is obvious. In the full loss cases, the benefit of increased field strength is also clear. Confinement is improved by operation at higher plasma currents permitted by the increased central field even though ignition is not attained. Thus the alpha particles provide a successively larger fraction of the total plasma losses as B_T increases. The uncertainty in the scaling laws used suggests a range of eventualities.

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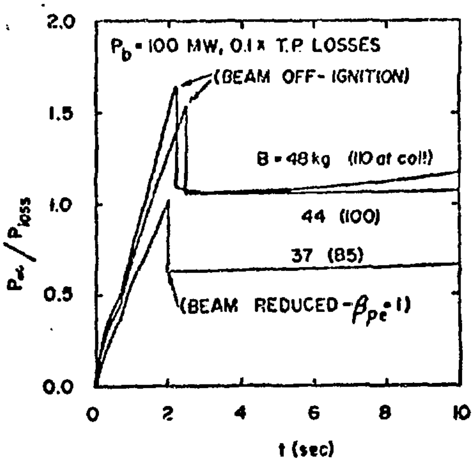

Fig. 1. P_α/P_{loss} ratio for reduced trapped-particle losses as a function of time for three values of toroidal field.

At the one end of the range, ignition occurs and the system produces ∿ 200 MW of fusion power under the set of assumptions used here; 50 MW of injection is required for heating. Even in this case, higher power operation is possible and depends upon detailed assumptions.

At the other end of the range, with less favorable containment properties, the system can be injection driven using 100 MW of 200 keV beams to produce 300 MW of fusion power of which 210 MW comes from background fusion events, and 90 MW comes from beam fusion events, yielding a total of 400 MW of recoverable heat.

In either the ignition of driven case the reference system provides reactor grade plasma operation and significant neutron wall loadings, and represents a suitable post TFTR step in the program. A more detailed statement of the plasma engineering work described here appears in a companion paper in this conference proceeding.[4]

Nuclear System

Blanket Criteria

The blanket for the EPR was designed to perform three functions:

1) absorb at least 90% of the plasma energy at the level of a few hundred MW, and convert it to heat at a temperature of 550°C,

2) (with the shield) attenuate the plasma neutrons and radiation to protect the winding of the toroidal field coils, and

3) provide some experimental modules where breeding of tritium could be demonstrated.

Blanket Design

The blanket is composed of sixty autonomous segments which, when clamped and welded together, make up the torus. In addition to the three primary functions performed by the blanket, it also constitutes the vacuum enclosure for the plasma.

Each blanket segment is composed of three compartments. The first compartment is the main absorber. All the blanket segments are identical except for the substitution of lithium for potassium in the experimental breeding segments. Potassium and lithium were selected as the absorbers because liquid metal absorbers are easily cooled themselves and good thermal conductivity is required to conduct heat from the first wall to the coolant tubes. Lithium is used in the experimental breeding module to produce tritium.

The second compartment is the reflector. A fraction of the relatively high energy neutrons which have passed through the liquid metal absorber is reflected back into the metal. This is not important for the non-breeding modules, but it is an essential feature of the tritium breeding test module. The breeding ratio in one of the experimental breeding modules is 1.2:1. Graphite was selected for the reflector because the reflected moderated neutrons reentering the absorber region are more effectively captured by the lithium-6 when reflected by graphite.

The third blanket compartment is a gamma shield composed of three concentric cylinders of stainless steel. In the annuli between these cylinders is liquid metal. The purpose of the liquid metal is to conduct and convect the heat from the stainless steel cylinders to the cooling tubes. Liquid potassium or lithium was selected as the heat transfer medium because the heat from the stainless steel gamma absorber is easily conducted through the liquid metal to the coolant tubes.

Calculations were performed for a driven system using neutral beam power and a system that ignites. Under steady-state conditions, the beam-plasma fusion power is 90 MW, and the Maxwellian fusion power is 210 MW. The neutron wall loading is ∿ 0.4 MW/m^2. The total recoverable power from this system is ∿ 400 MW(th).

The first structural wall of the blanket is to be protected from the plasma bombardment by a separate inner wall. The heat from this inner wall will be radiated to the first structural wall of the blanket. The blanket cooling has been designed to remove this heat as well as that produced by particle absorption in the blanket structure and absorber metal. The temperature of this first wall will be about 1400°C.

Materials

Materials for use in advanced CTR systems must be chosen to provide maximum reliability of the reactor while providing minimum restriction on the reactor design and operation.

The first structural wall of the reference design is type 316 stainless steel. This material is available, economical, and will be adequate to meet the needs of EPR.

The first radiation wall presents a more difficult material choice. While it appears likely that a bare type 316 stainless steel is not the optimum plasma-structure interface, the properties required by the interface material are not well defined. A tentative choice has been made for the reference design, namely, a "shingle" wall, with the shingles held in place by a refractory metal lattice attached to the first structural wall. Possible shingle materials include W, Nb, C, refractory carbides and refractory oxides.

Neutronics

The total fraction of neutron and gamma ray energy deposited in the blanket assembly is 89.8%, which is consistent with the design specification for 90% energy coupled winding provides the shielding function and in addition serves to produce the vertical field. The closeness of this winding to the plasma allows the use of a decay index $n \equiv R/B_r \, \partial B_r/\partial R = 0.5$, which gives good stability against vertical plasma displacements. A further set of external vertical trim coils is provided for extra flexibility. In principle, this system is capable of shielding the toroidal field coils completely from the OH field, and in practice it can come close to achieving this. First estimates using the reference design geometry indicate a factor of at least 3.3 reduction in the poloidal field level at the toroidal field coils. Optimization is under way and further improvement of the shielding effectiveness is expected. Since the rate of the TF coil design, the factor of 3 reduction is already significant.

In order to achieve reasonable currents and voltages, reference coil parameters are:

- superconducting primary-OH system of 400 turns producing 132 volt-seconds
- resistive shield-VF winding of 16 turns at cryoresistive temperatures
- superconducting decoupling coils of 16 turns
- superconducting VF-trim coils.

The resolution of the electrical energy storage, switching, impedance matching and control problems is made unclear because of the many uncertain trade-offs potentially available. The voltage and current capacity limitations of the superconducting coils combined with the cost and difficulty of high energy, high power storage and switchgear produce a situation whose nominal reference design solution is still under active consideration. For a base case, the following system has been chosen.

The air core magnetic circuit is composed of twelve physically separate, magnetically coupled coils energized by a large, multi-component homopolar generator whose design is based upon extrapolations from present day devices.[11] In this scheme, the demands placed upon switchgear are high. Alternate ways of providing the switching and improving the paralleling of coils are being pursued.

Neutral Beam Injection

The design goal for neutral beam power delivered to the EPR plasma is 100 MW at 200 keV.

The duration of a beam pulse is determined by a) both the volt-second OH limit if the EPR is capable of puring helium and refueling, and the impurity-buildup limit, and b) whether or not the plasma ignites.

The reference design system is taken to be a negative ion source, with no recovery of charged particle fraction energy. This design couples the ORNL beam-handling experience gained from the 150 keV TFTR beams with the development of a suitable negative ion source to give an efficient overall system. The benefits gained from a direct ion energy recoverer do not yet warrant the added complexity of the recovery system.

The backup design is taken to be a positive ion source with no recovery of the energy in the charged particle fraction. That design is a direct extension of the 150 keV TFTR beams and injectors to slightly higher voltage and to twice the current per injector. All ORNL TFTR experience is directly applicable here.

The more complex negative ion reference scheme requires the development of suitable negative ion sources, but requires lower input electrical power. That suggests its use for a steady-state, beam-driven EPR. The positive ion backup scheme is a direct extension of the TFTR injectors, requiring less development, but requiring more input electrical power. This suggests its use for an igniting EPR, with an energy storage system supplying this startup power.

The two systems, reference design and backup design, are compared in Table 3. The beam transport efficiencies η_1, η_2 and η_4 represent goals for the beam development program to achieve by the time needed for the EPR. All parts of the reference design system except the negative ion source are taken directly and scaled up in performance where needed from the ORNL beam system design for TFTR.[12] The negative ion sources must still be developed; the backup design beam system may be taken directly from the ORNL TFTR beam system design.

Table 3. Comparison of Reference Design and Backup Design

	Reference Design (D^-)	Backup Design (D^+)
I_o (amps)	500	500
η_1 useful extraction eff.	0.95	0.95
η_2 grid and geometric eff.	0.90	0.90
η_3 neutralizer eff.	0.90	0.90
η_4 drift tube eff.	0.97	0.97
F_o = conversion eff. of equilibrium cell	0.67	0.20
I_A (amps)	1000	3333
P_A (MW)	200	666
Number of source	12	36
I_A/ source	83	93
Beam lines	6	6
Sources/line	2	6
P_o (MW)	100	100
Efficiency P_o/P_A	0.50	0.15

Tritium Handling Systems

The tritium handling systems for EPR consist of the main and injector vacuum systems, tritium process systems, and tritium containment and atmosphere clean-up systems.

Vacuum System

The pumping speed requirements for the main vacuum system have not been determined. The criteria for allowable neutral density outside the plasma are dependent upon energy loss and sputtering problems caused by charge exchange between neutrals and hot ions. For the reference design a nominal pumping speed of 1,500,000 liters per second was chosen based on a gas of mass 5 (DT) at 550°K. This speed can be achieved with 40 vacuum ports of 1 m diam, 5 m length, each being pumped by a 1 m diameter cryosorption pump. If 1.5×10^{22} ions per second leave the plasma, the equilibrium density of neutral gas outside the plasma would be 1×10^{13} atoms/cm^3.

Tritium Process System

Tritium process systems include equipment for:

1. removing D-T mixtures from the main vacuum systems,
2. purifying the D-T mixture for recycle to feed systems,
3. separating hydrogen isotopic impurities from D-T mixtures,
4. plasma fueling and injector feed systems,
5. tritium storage systems,

deposition in the blanket.[5] The graphite layer and the shield absorb 10.1% of the total heat and the remainder (<< 0.1%) is absorbed in the TF-coil assembly.

Estimates of the radiation damage in the first iron wall and in the first copper winding in the TF coil are given in Table 2.

Table 2. Estimates of Radiation Damage in First Iron Wall and in Toroidal Field Coil[a]

	Gas Production	
	Hydrogen Atoms	Helium Atoms
dpa/year	atoms/year	atoms/year
	First Iron Wall	
1.89	4.73 × 10[illegible]	1.47 × 10[illegible]
	First Copper Winding in TF Coil	
4.20 × 10[illegible]	3.75 × 10[illegible]	8.71 × 10[illegible]

a. For assumed wall loading of 0.168 MW/m^2.

If the radiation damage data for 0.168 MW/m^2 wall loading are extrapolated to a wall loading of 1.0 MW/m^2, the atomic-displacement rate is 11.25 dpa/y, a value that is consistent with the rates predicted by Kulcinski, Doran, and Abdou[6] and by Williams, Santoro, and Gabriel.[7] A more detailed discussion of the nuclear systems in the ORNL EPR is contained in References 4 and 8.

Magnetic Field System

A fundamental characteristic of the entire magnetics system is the maximum toroidal field strength, set in the reference design to be 11 T. This value is presently judged to be the highest one feasible in the EPR design and development time frame.

Toroidal Field System

There are several constraints on the toroidal field coils which dictate rather specifically some of the design choices.

- For example, the field at the windings of the of the toroidal field coils will be 11 T to provide the reference design field of 4.8 T at the plasma centerline. Therefore, the design uses Nb_3Sn as superconductor in the higher field regions in addition to the more ductile NbTi for those parts of the coil at 7 T or less.

- Even though the TF coils are in principle shielded from pulsed OH fields, it is likely that imperfections in the shielding current arrangement and the equilibrium fields will result in rates of field change of a few tenths of 1 T/sec; therefore, filamentary conductors capable of handling these variations are specified rather than tapes.

- Use of non-superconducting coils is precluded because of the very large power requirements.

- It is required that if a normal region is produced by some transient—conductor slippage, plasma instability, etc.,—then the coil should recover, and in a predictable fashion. For the required current density, field strength, and coil dimensions, aluminum is a better choice than copper as the stabilizing material, and forced flow cooling through a well-defined passage is essential. We have, therefore, chosen to specify a cabled conductor inside a hollow steel tube. The tube walls are substantial and the electrical insulation is placed between the cable and the tube, allowing the tubes to be metallurgically joined to each other and to the case. An integral structure is thus formed in which each turn is separately and securely fixed. This arrangement also spreads the conductors over a larger cross section of the coil, giving a higher on-axis field for a given maximum field at the conductor. The outer case enveloping the coil provides a secondary leak-tight enclosure and efficient restraint against bending.

- For such a large coil it is desirable to grade the conductor according to the field in which it will be used, and to minimize the number of conductor joints. This leads not to a pancake coil, but to a layer winding, wherein the maximum field along a given conductor on any layer is nearly constant. In the layer winding, grading is thus accomplished without internal joints, i.e., all joints occur at the ends of each layer, at the end of the coil. Furthermore, in contrast with large monolithic conductors, the large cable conductors can be supplied in very long lengths, and joints between layers are unnecessary unless it is desired to change the conductor at that point. However, for sufficient cooling each layer must have a helium inlet and outlet.

The considerations listed above indicate a coil whose cross section and principal parameters are shown in Fig. 2. Further discussion of the EPR TF coil design is to be found in Reference 9.

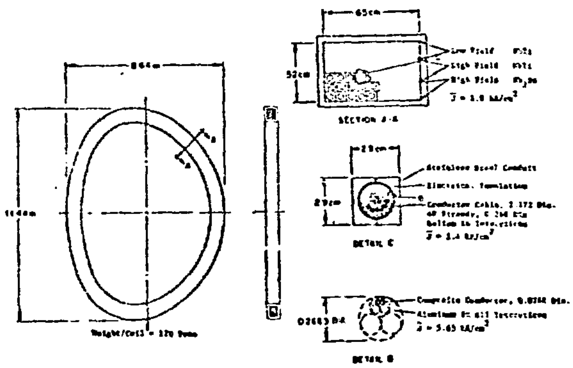

Fig. 2. EPR reference design toroidal field coil.

Poloidal Magnetic Field

The function of the poloidal system is to create the plasma current, and to maintain the plasma column in equilibrium by suitably shaped vertical fields. The design gives special emphasis to reducing stored magnetic energy, and to shielding the superconducting toroidal field coils from pulsed magnetic fields. The electromagnetic shielding is a vital function, since pulsed fields can quench the superconducting windings. Pulsed fields come from both the OH winding and from the plasma current and may especially important if the plasma exhibits disruptive instabilities, a universal feature of present tokamaks, which cause rapid current changes and large induced voltages.

The design used here is a version of the "STATIC" system recently proposed by F. B. Marcus.[10] In the STATIC system, the poloidal magnetics include an air core transformer in a special arrangement. This arrangement both requires much less energy than the usual air core transformer, in which a poorly coupled winding is outside the toroidal field coils, and does so in such a way that it shields the toroidal windings from being immersed in poloidal flux. A closely

6. tritium recovery from blanket modules,
7. preparation of tritium-containing wastes for disposal

A schematic flow chart of the reference design tritium processing system is shown in Fig. 3.

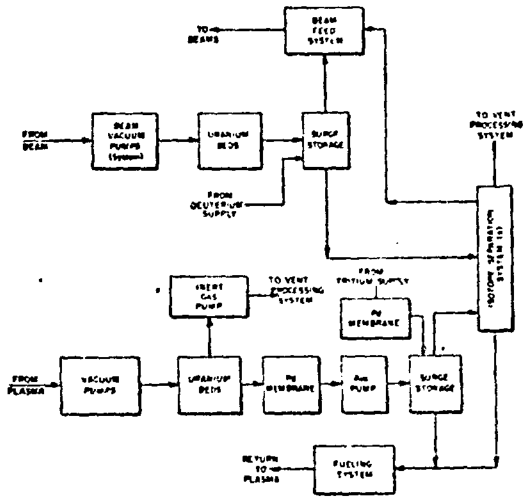

Fig. 3 Schematic flow chart of EPR reference design tritium processing system.

The blanket recovery system is one of the least well understood items in the tritium processing area, and considerable experimental work will be required to determine the most promising system and to provide the necessary design data. Lithium is currently considered the most promising breeding material. A small side stream can be removed for processing; we are currently considering extraction, permeation, and sorption processes for tritium recovery.

Tritium Containment

Because of the huge quantities of tritium which will be handled in the fuel cycle and the low permissible release values, it will be necessary that all components containing significant quantities of tritium be doubly contained. Techniques to achieve double containment will include glovebox-type enclosures over components such as pumps which may require periodic maintenance and double-walled pipes. The atmospheres of enclosures and the annuli in the double-wall pipes will be maintained at a slight vacuum with respect to the reactor building and must be processed through atmosphere cleanup systems to remove tritium and thus reduce release to the reactor building. The atmosphere cleanup systems will consist of heaters, recombiners using palladium or copper oxide, chillers and regenerable molecular-sieve drier beds.

The reactor building will contain all of the major tritium-handling components. The basic design of the reactor building atmosphere-cleanup system will be similar to, but larger than, the small enclosure atmosphere-cleanup systems.

Summary

Figure 4 shows the ORNL EPR reference design as an assembly of the individual systems described above. Preliminary consideration of the design has included fundamental questions of assembly, disassembly and remote maintenance and an examination of the research and development needed to support the implementation of the design. Continuing effort is being placed upon refinement of this design and understanding of the requirements for research, development, fabrication and installation.

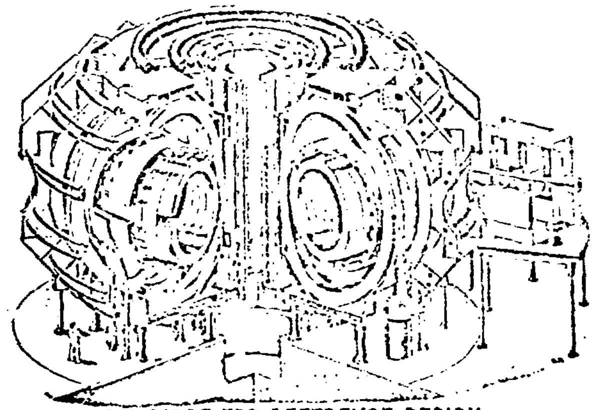

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Figure 4

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