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Oak Ridge Tokamak Experimental Power
**Oak Ridge Tokamak Experimental Power
Reactor Study—1976**

**Part 5
Engineering**

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Fusion Energy Division

OAK RIDGE TOKAMAK EXPERIMENTAL POWER REACTOR STUDY

Part 5

ENGINEERING

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The other reports in the Oak Ridge Tokamak Experimental Power Reactor Study — 1976 are:

ORNL/TM-5572	Part 1. EPR Summary
ORNL/TM-5573	Part 2. Plasma Engineering
ORNL/TM-5574	Part 3. Magnet Systems
ORNL/TM-5575	Part 4. Nuclear Engineering
ORNL/TM-5577	Part 6. Research, Development, and Demonstration Needs

ABSTRACT

The Experimental Power Reactor (EPR) studies which have been performed at Oak Ridge National Laboratory (ORNL), at Argonne National Laboratory (ANL), and at General Atomic Company (GAC) during the last two years have investigated the design and development aspects associated with the production of a large tokamak to be operated in the mid-1980's and to be the precursor of a fusion Demonstration Reactor Plant (DRP).

At ORNL, the first year was devoted to scoping studies. At the end of this period, a reference concept was selected. During the past year, the reference concept has been pursued in more depth; based on these more detailed investigations, major research and development needs have been identified and documented.

The results of the past year's effort in the engineering areas are documented in this report. The discussion covers system identification and overall description, power conversion, plant layout, structural design, remote maintenance and assembly, program planning, and schedules. Five companion reports being issued as parts of this composite report present information in the other discipline areas.

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FOREWORD

The Division of Magnetic Fusion Energy (DMFE) within the Energy Research and Development Administration (ERDA) identifies in the fusion development program for tokamak power reactors progressively larger confinement systems leading to a Demonstration Reactor Plant (DRP) by the end of this century. In this program, the step after the Tokamak Fusion Test Reactor (TFTR) and before the DRP is an Experimental Power Reactor (EPR), with operation scheduled for the mid-to-late 1980's.

In FY 75, scoping studies were performed, devoted primarily to determining functional and technological considerations associated with the design of an EPR. A reference concept was selected and during FY 76 this concept has been studied in more depth. From this detailed examination of a particular concept, a more realistic assessment has been made of the system and discipline requirements for an EPR and of the constraints imposed on the design by current theoretical understanding, existing technology, and near-term development.

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1. INTRODUCTION

Following the preliminary EPR Reference Design work in CY 75,¹ a program plan was formulated to begin the conceptual design of a power-producing EPR to be operational in the mid-1980's. However, due to insufficient funds in FY 76, followed by an additional cut in the President's anticipated budget, it was decided in the late fall that the normal conceptual design activities for a large ERDA capital project² could not begin. The consensus was to concentrate the engineering effort on a few select systems of the reference design which represented areas of major uncertainty.

The primary systems included in this year's engineering effort were:

- blanket and shield
- tokamak mechanical structure
- toroidal magnets
- poloidal field
- remote maintenance and assembly
- reactor plant design
- program plan.

While the overall size and shape of the tokamak device have not been changed, several components and design concepts have changed extensively as a more detailed study revealed improvements to the original concept.

This part of the composite report presents an overall description of the updated reference design, followed by a detailed summary of the system investigations. The engineering design details for the blanket and shield and for the field coil systems are reported in Part 3 (ORNL/IM-5574) and Part 4 (ORNL/IM-5575) of the composite report. The plasma physics investigations, including operations and performance criteria, are reported in Part 2 (ORNL/TM-5573).

2. SUMMARY

The engineering studies were based on an EPR with a nominal thermal output of 410 MW. The design point was selected as 800 MW(t), which is considered as an optimistic upper limit. The revised reactor design is illustrated in Figs. 2.1 and 2.2, and a first attempt to define the overall plant layout is shown in Fig. 2.3.

The principal accomplishments of the engineering work are as follows:

1. A system breakdown and identification were specified to provide a basis for project planning and control.
2. Design and operating parameters were tabulated for the major reactor systems.
3. An operations cycle chart was constructed.
4. An EPR plant concept was identified considering both reactor and support facilities.
5. A power conversion system and resulting heat balance were determined.
6. A program was outlined for development of structural design criteria.
7. The structural analysis reported in the EPR scoping study for FY 75 was extended to include the effects of out-of-plane loads.
8. Structural design alternatives were evaluated, and the continuous toroidal field (TF) coil support system is recommended.
9. A computer technique for simplified finite element structural analysis was verified.
10. The effect of remote maintenance on plant availability and the related electrical cost were determined.
11. The requirement of remote maintenance was considered in the mechanical design and assembly of the EPR.
12. The remote maintenance requirements were categorized into six areas for recommended R and D programs.
13. A preliminary engineering job plan was established for an FY 79 "Line Item" EPR project.
14. Progressive Events Charts were prepared to outline the necessary steps from the study phase through construction of an EPR.

Fig. 2.1. EPR elevation view.

① AIR CORE PRIMARY WINDINGS	⑤ T.F. COILS (CONDUCTOR, BOBBIN, DEWAR)
② DECOUPLING WINDINGS	⑥ SHIELD
③ V.F. TRIM WINDINGS	⑦ BLANKET
④ SHIELDING/V.F. WINDINGS	⑧ VACUUM CONTAINMENT

Fig. 2.2. EPR plan view.

Fig. 2.3. EPR plant layout.

3. REFERENCE DESIGN DESCRIPTION

The EPR device is a tokamak using 20 superconducting magnets to maintain the toroidal field. These magnets are arranged on a stainless steel structure to form a torus, with the noses of the coils contacting a cylindrical center bucking column. The coils are a modified oval shape. The torus formed by the coils has a major radius of 6.75 m.

Inside the TF coils there is a circular cross section water-cooled vacuum vessel 4.5 m in diameter. Surrounding this vacuum wall there is a blanket 0.5 m thick. The blanket and the water wall are in the shape of a circular wedge. Sixty of these units are required to complete the torus which is a 60-sided polygon.

Outside the blanket there is a void space 30 cm thick and a water-cooled shield about 0.5 m thick. In recesses on the inner surface of the shield there are 16 water-cooled copper coils, each of which completely encircles the plasma, wall, and blanket. The shield serves as the mechanical support for these coils and for the blanket assemblies as well. The shield segments rest on tripod support columns which in turn rest on the machine foundation pad.

There are three types of superconducting poloidal field windings. These are the ohmic heating, vertical field trim, and decoupling windings. Of these, parts of the ohmic heating and decoupling windings are located inside the center bucking column of the structure. Parts of these windings and all of the vertical field trim windings are located outside the TF coils, and their mounting structure connects to the TF coils.

There are two vacuum ports in the space between each TF coil. The neutral beam injection ports are also located between the coils. These are the largest penetrations into the plasma and require large shadow shields to prevent excessive radiation leakage through the openings. Other diagnostic openings are located at various points around the torus, but these are assumed small enough that they do not present a serious problem as far as radiation leakage is concerned.

While the size and general configuration of the EPR have changed little from the reference design of 1975, certain components have undergone extensive re-design and these new designs will be described in detail in this and other sections of the report.

3.1 Systems Breakdown

As an aid in the overall management of the EPR project, a systems breakdown is proposed. The breakdown consists of nine major categories, which include 29 systems. System descriptions are used as the basis for control of reactor design projects, as recommended by RDT Standards.³ The recommended EPR systems breakdown is listed in Table 3.1.

Table 3.1 EPR systems breakdown

- (1) Overall Plant
 - 1.0 EPR Overall System Description
- (2) Tokamak Reactor Device
 - 2.0 Blanket and Shield System
 - 2.1 Toroidal Magnets System
 - 2.2 Poloidal Field System
 - 2.3 Structural System
 - 2.4 Impurity Control System

- (3) Electrical Energy Input
 - 3.0 Electrical Energy Supply and Distribution Systems
 - 3.1 Electrical Energy Storage and Pulse Systems
- (4) Instrumentation
 - 4.0 Instrumentation and Control Systems
 - 4.1 Diagnostics Systems
- (5) Heating
 - 5.0 Plasma Heating System
- (6) Tritium-related
 - 6.0 Tritium Processing System
 - 6.1 Tritium Storage System
 - 6.2 Plasma Fueling System
 - 6.3 Vacuum Pumping System
 - 6.4 Lithium Processing System
- (7) Thermally-related
 - 7.0 Primary Coolant System
 - 7.1 Secondary Coolant System
 - 7.2 Heat Conversion/Electrical Generation System
 - 7.3 Waste Heat Processing System
 - 7.4 Cryogenic Cooling System
- (8) Auxiliary
 - 8.0 Reactor Building Complex
 - 8.1 Maintenance and Assembly System
 - 8.2 Safety and Fire Protection System
 - 8.3 Liquid and Gas Storage System
 - 8.4 Balance of Plant Systems
 - 8.5 Site Criteria/Improvement
- (9) Environmental Protection
 - 9.0 Tritium Containment System
 - 9.1 Radioactive Waste Handling System

3.2 Major Systems Description

A brief description of the major systems of the current EPR device and the associated design and operating parameters for each of these systems are listed in the following sections. These systems are identified on Fig. 3.1. The six systems are:

1. Overall system
2. Blanket and shield system
3. Toroidal magnets system
4. Poloidal field system
5. Plasma heating system
6. Vacuum pumping system

3.2.1 Overall System

The EPR is a DT-burning air core tokamak with superconducting coils. The duty cycle is high (87%), with a burn time of 100 sec. The machine has a major radius of 6.75 m and a circular plasma radius of 2.25 m (aspect ratio of 3.0). A blanket is provided which transfers 410 MW(t) from the plasma to a steam power cycle. A neutral beam system is provided to inject 50 MW of power into the plasma to achieve ignition.

The nuclear design of the EPR is based on a neutron wall loading of 1 MW/m^2 , which is equivalent to a power output of 800 MW(t). Operation at this power level is considered an optimistic upper limit.

Table 3.2 gives a listing of the overall system parameters of the EPR.

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Fig. 3.1. Major EPR systems:

1. Overall system.
2. Blanket and shield system.
3. Toroidal magnets system.
4. Poloidal field system.
5. Plasma heating system.
6. Vacuum pumping system.

Table 3.2. Overall system parameters

A. General Characteristics1. Requirements

	<u>Symbol</u>	<u>Value</u>
Device lifetime	t_L	10 years
Duty cycle	dc	87%
Availability - normal	--	80%
Thermonuclear power (ignition)	P_O	410 MW

2. Other Features

Plasma radius	a	2.25 m
Major radius	R_O	6.75 m
Aspect ratio	A	3.0
Plasma edge to winding distance	Δ	1.40 m
Plasma volume	V	675 m ³
Overall height	H	15 m
Overall diameter	D	23.3 m

B. Plasma Parameters

Burn time	t_B	100 sec
Current	I_p	7.2 MA
$N_e \tau_E$	$N_e \tau_E$	2.7×10^{14} sec cm ⁻³
Plasma temperature		
Ion temperature	T_i	12.2 keV
Electron temperature	T_e	13.0 keV
Beta poloidal	β_p	1.9
Safety factor	q	2.5
Electron density	N_e	7.4×10^{13} cm ⁻³
Fuel ion density	N_i	7.0×10^{13} cm ⁻³
Impurity level (t=50 sec)	Z_{eff}	1.34
Injection power into plasma	P_b	50 MW
Injected deuteron energy	E_b	200 keV
Particle confinement time	τ_p	9.6 sec
Energy confinement time	τ_E	3.7 sec

C. Electromagnetic Parameters

Maximum toroidal field	B_{max}	11 T
Toroidal field on axis	B_T	4.8 T
Number toroidal coils	N_c	20
Magnetic field ripple at plasma edge	ΔB	2.2%
Conductor configuration	--	composite cable in a square, aluminum alloy conduit
Superconductor in the high field region	--	Nb ₃ Sn in Cu-Sn matrix conductor

in the moderate and low field region	--	NbTi in Cu matrix conductor
Avg. poloidal field at plasma edge	B_p	0.64 T
Poloidal field coil core type	--	air
Poloidal coils volt-seconds	VS	165 Wb

D. Thermal

Blanket maximum power rating	P_R	800 MW
Blanket coolant	--	helium, water
Blanket coolant temperature inlet	T_{in}	204 °C
outlet	T_{out}	370 °C
Blanket coolant pressure	P_{He}	70 atm
Shield coolant	--	borated water
Coil coolant (superconducting)	--	helium

E. Nuclear Parameters

Neutron wall loading (operating point)	P_w	0.55 MW/m ²
Neutron wall loading (design point)	P_w	1.07 MW/m ²
Neutron flux on first wall (operating point)	ϕ_w	2.4×10^{13} n/cm ² /sec
Neutron flux on first wall (design point)	ϕ_w	4.7×10^{13} n/cm ² /sec
Tritium breeding ratio (in one experimental module)	BR	1.15

3.2.2 Blanket and Shield

The primary purposes of the blanket and shield are to absorb the plasma energy and convert it to heat, to attenuate the plasma neutrons and radiation, to provide for breeding of tritium, and to form the vacuum enclosure for the plasma. The blanket is composed of 60 segments which when assembled make up the torus. Each segment is further divided into 12 modules which constitute the basic building blocks of the blanket. The first wall of the blanket, which receives the alpha particle heat, is water-cooled. The liquid metal absorber portion of the blanket, which receives the high energy neutrons from the plasma, is cooled by helium.

The shield is composed of 20 stainless steel and lead segments fitted inside the TF coils. The shield is cooled by borated water and forms a thermal barrier between the high temperature blanket and the cryogenic TF coils. Neutron flux coming through the blanket is attenuated by a factor of 2045 in the shield.

Table 3.3 gives the blanket and shield system parameters.

Table 3.3. Blanket and shield system parameters

A. General Characteristics1. Requirements

	<u>Symbol</u>	<u>Value</u>
Life	t_L	2-5 years
Duty cycle	dc	100%
Availability - normal	--	98%

2. Other Features

Number of segments comprising torus	--	60
Segment cross-sectional shape	--	circular
Number modulus/segment	--	12
Blanket inside radius	BK_{IR}	2.25 m
Shield outside radius	S_{OR}	3.55 m
Blanket thickness	Th_B	0.515 m
Shield thickness	Th_S	0.485 m

B. Mechanical Parameters

Structural material	--	316 SS
---------------------	----	--------

C. Thermal Parameters

Design point heat flux on first wall (alpha heat only)	q_W	0.27 MW/m ²
First wall coolant	--	water
Water inlet temperature	T_{in}	38 °C
Water exit temperature	T_{out}	121 °C
Maximum temperature structural wall	T_s	536 °C
Blanket absorber (non-breeding modules)	--	potassium +316 SS
Blanket absorber (breeding modules)	--	lithium
Maximum temperature, lithium	T_L	573 °C
Blanket reflector material	--	graphite
Maximum temperature, reflector	T_R	557 °C
Blanket coolant	--	helium
Helium flow rate (design)	$\dot{m}$	800 kg/sec
Helium inlet temperature	T_{in}	204 °C
Helium outlet temperature	T_{out}	370 °C
Helium inlet pressure	P_{in}	70 atm
Shield coolant	--	borated water

D. Nuclear Parameters

First wall neutron loading (operating)	P_W	0.55 MW/m ²
First wall neutron loading (design)	P_{WD}	1.07 MW/m ²
Neutron flux on first wall (operating)	ϕ_W	2.4×10^{13} n/cm ² /sec
Neutron flux on first wall (design)	ϕ_W	4.7×10^{13} n/cm ² /sec
Tritium breeding ratio (each experimental module)	BR	1.15
Neutron flux attenuation (blanket)	F_B	factor of 26

	<u>Symbol</u>	<u>Value</u>
Neutron flux attenuation (shield)	F_S	factor of 2045
Neutron flux attenuation (total)	F_T	factor of 53,000
Shield material	--	H ₂ O + 316 SS & lead slab
Radiation damage in first structural wall	D_R	11 DPA/yr
Hydrogen atoms produced in first structural wall	A_H	424 appm/yr
Helium atoms produced in first structural wall	A_{He}	129 appm/yr

3.2.3 Toroidal Magnet System

The toroidal magnetic field generated by the toroidal magnet system is combined with the poloidal magnetic field generated by the plasma current to produce spiral magnetic field lines in the plasma. The toroidal field is produced by 20 oval-shaped, superconducting, forced-coolant-flow field coils with a minor bore diameter of 7.4 m and a major bore diameter of 9.7 m. These coils produce a maximum field at the winding of 11 T and an on-axis field of 4.8 T at the plasma.

The superconducting elements are composite wires composed of Nb₃Sn filaments in a CuSn matrix in the high field area and NbTi filaments in a copper matrix in the lower field portion of the coil (<6.5 T). The conductor coolant is helium.

Table 3.4 gives the toroidal magnet system parameters.

Table 3.4 Toroidal magnet system parameters

A. General Characteristics

1. Requirements

	<u>Symbol</u>	<u>Value</u>
Magnetic field on plasma axis	B_T	4.8 T
Duty cycle	dc	100%
Magnetic field ripple at plasma	ΔB	2.2%
Life	t_L	10 years
Maximum overall ratio of magnet operating power to reactor thermal output	$P_{TF/O}$	10%

2. Other Features

Horizontal opening	d_{tf}	7.4 m
Vertical opening	h_{tf}	9.7 m
Major radius of tokamak	R	6.75 m
Number of coils	N_C	20
Coil shape	--	oval
Weight/coil	W_{coil}	85,000 kg
Access space between coils - maximum	S_{max}	2.62 m
Access space between coils - 45° angle	S_{45}	2.35 m

B. Mechanical Parameters

Strain in conductor matrix	ϵ_m	<0.2%
Centering force/coil	F_C	250 MN
Structural material	--	300 series mod. SS

	<u>Symbol</u>	<u>Value</u>
<u>C. Electromagnetic Parameters</u>		
Maximum field at winding	$B_{\max}$	11 T
Ampere turns/coil	A-T	8.1 MAT
Stored energy/coil	W_c	1.45 GJ
Conductor (cable) current	I_c	19.5 kA
Current per turn (4 conductors)	I_t	78 kA
Self-inductance	H	9.5 H
Conductor type	--	multifilament superconductor
Conductor material (in high field)	--	Nb ₃ Sn superconductor with CuSn matrix plus a Cu or Al wire
Critical current density in Nb ₃ Sn at 11 T, 6.0 K	$J_{s.c.}$	139 kA/cm ²
Critical current density in NbTi at 6.5 T, 4.0 K	$J_{s.c.}$	112 kA/cm ²
Current density averaged over cable (including helium space)	J_{cable}	6.21 kA/cm ²
Current density averaged over winding (cable, helium, and aluminum-alloy conduit)	J_w	4.03 kA/cm ²
Current density averaged over windings plus coil case	J_{AC}	1.79 kA/cm ²
Conductor material (in low field)	--	NbTi superconductor with Cu matrix plus a Cu or Al wire
<u>D. Thermal Parameters</u>		
Conductor coolant	--	supercritical helium
Average helium inlet temperature	T_{in}	4.0 K
Average helium inlet pressure	P_{in}	5.0 atm
Average steady-state temperature rise	ΔT_{He}	0.93 K
Helium flow rate/coil	$\dot{m}$	140 g/sec
Total refrigeration required at 4.0 K (includes pumping power)	P_{Refr}	25.5 kW
<u>E. Nuclear Parameters</u>		
Average neutron heating/coil (design point)	W_n	800 W
Neutron flux at coil (design point)	ϕ_c	4.9×10^9 n/cm ² /sec
Neutron fluence/yr (E > 0.1 MeV) (design point)	--	1.55×10^{17} n/cm ²

3.2.4 Poloidal Magnet System

The purposes of the poloidal magnetic system are (1) to induce a current in the plasma, (2) to protect the superconducting toroidal field coils from magnetic flux changes, and (3) to maintain the plasma column in equilibrium. The poloidal magnetic system is composed of the ohmic heating coil, the shield-vertical field coil, the decoupling coil, and the trim-VF coil.

The ohmic heating coil is primarily a superconducting central air core solenoid. This coil swings from full positive current to full negative current (with intermediate current storage in a homopolar generator) in 2 sec. This creates a flux change of 110 Wb in the central solenoid which is the primary driving force inducing a current of 7.2 MA in the plasma.

The shield-vertical field coil is composed of 16 water-cooled windings which are located between the blanket and shield and run in a toroidal direction around the plasma. The plasma current induces an equal and opposite current in these windings which essentially cancels flux pulses at the toroidal field coils. Also, these windings create the major portion of the vertical field which maintains the plasma column in equilibrium.

The decoupling coil is superconducting and is composed of 16 turns. This coil is in opposite sense to the ohmic heating coil and is connected in series to the shield-vertical field coil to cancel out induced voltage from the ohmic heating coil to the shield-vertical field coil.

The trim coil is superconducting and is composed of 16 windings. These windings are outside the toroidal field coils and create an auxiliary vertical field to supplement the vertical field generated by the shield-vertical field coil. This correction supplied by the trim coil is necessary since the shield-vertical field coil is constrained to carry a current equal and opposite to the plasma current for shielding purposes, and in so doing, sacrifices stabilizing flexibility.

Table 3.5 gives the poloidal magnet system parameters.

Table 3.5. Poloidal field system parameters

I. Primary OH Coil

A. General Characteristics

1. Requirements

	<u>Symbol</u>	<u>Value</u>
Life	t_L	10 years
Duty cycle	dc	100%
Max. overall ratio of magnet operating power to reactor thermal output	$P_{pf/o}$	10%
Induced plasma current	I_p	7.2 MA

2. Other

Coil shape/type	--	a long superconducting central air core solenoid with outer coils to return flux outside the TF coil
Inside radius of solenoid	R_I	1.6 m
Outside radius of solenoid	R_{os}	1.95 m
Length of central core	L_s	4.5 m
Number of turns (total)	N_{oh}	1220

B. Mechanical Parameters

Average hoop stress	σ_s	12,500 psi
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C. Electromagnetic Parameters

Ampere turns	A-T	61 MAT
Max. field at winding	B_{smax}	7.2 T
Field at coil axis	B_{sax}	7.1 T

	Symbol	Value
Max. rate of flux change	$\dot{B}$	7.2 T/sec
Coil swing time (full forward to full reverse current)	t_s	2 sec
Coil volt-seconds	VS	110 Wb
Stored energy	W	1.9 GJ
Coil voltage (max.)	V_c	35 kV
Max. coil current (total)	I_c	100 kA
Coil electrical linkage	--	coil is divided into 2 parallel groups, each consisting of 610 turns
Current/group	I_G	50 kA
Conductor material	--	NbTi in a Cu-CuNi mixed matrix
Current density overall	J_{OA}	1.46 kA/cm ²
Current density in NbTi	J	35.4 kA/cm ²
Critical current density in NbTi (at 4.2 K, 7 T)	J_C	70 kA/cm ²
Self-inductance (single turn)	L_{oo}	1.03 μ H
Mutual inductance to plasma (single turn)	M_{op}	1.11 μ H
Mutual inductance to shield-VF coil (single turn)	M_{os}	1.12 μ H
Mutual inductance to trim coil (single turn)	M_{ot}	0.11 μ H
Mutual inductance to decoupling coil (single turn)	M_{od}	1.09 μ H

D. Thermal Parameters

Conductor coolant	--	supercritical helium
Helium inlet temperature	T_{in}	4.5 K
Helium inlet pressure	P_{in}	5.7 atm
Total refrigeration required at 4.5 K (includes pumping power)	P_{Refr}	15 kW

II. Shield-VF Coil

A. General Characteristics

1. Requirements

Life	t_L	10 years
Duty cycle	dc	87%

2. Other

Coil shape/type	--	coils surround torus at radius of 3.15 m from the plasma axis
Number of coils	N_{SH}	16

B. Electromagnetic Parameters

Ampere turns	A-T	7.2 MAT
Coil volt seconds	VS	55 Wb

	Symbol	Value
Stored energy	W	120 MJ
Max. shield coil current (total)	I_{sc}	7.2 MA
Current/coil	I_{ac}	450 kA
Conductor material	--	Cu
Self-inductance (single turn)	L_{ss}	12.3 μ H
Mutual inductance to plasma (single turn)	M_{sp}	9.7 μ H
Mutual inductance to OH (single turn)	M_{so}	1.12 μ H
Mutual inductance to trim coil (single turn)	M_{st}	3.06 μ H
Mutual inductance to decoupling coil (single turn)	M_{sd}	1.30 μ H
Induced vertical field at plasma	B_v	0.48 T

C. Thermal Parameters

Conductor coolant	--	water
Coolant inlet temperature	T_{in}	310 K
Coolant temperature rise	ΔT	55.5 K
Coolant flow rate (total)	$\dot{m}$	235 kg/sec

III. Decoupling Coil

A. General Characteristics

1. Requirements

Life	t_L	10 years
Duty cycle	dc	87%
Max. overall ratio of magnet operating power to reactor thermal output	$P_{dc/o}$	10%

2. Other

Coil shape/type	--	a superconducting solenoid concentric with, but in the opposite sense to, the primary OH coil and connected in series to the shield coils
Radius of solenoid	R_s	2.0 m
Length of central core	L_s	12 m
Number of turns	N_{dc}	16

B. Electromagnetic Parameters

Ampere turns	A-T	7.2 MAT
Max. decoupling coil current (total)	I_c	7.2 MA
Conductor material	--	NbTi in a Cu-CuNi mixed matrix
Self-inductance (single turn)	L_{dd}	1.59 μ H
Mutual inductance to plasma (single turn)	M_{dp}	1.35 μ H
Mutual inductance to OH coil (single turn)	M_{do}	1.09 μ H

	<u>Symbol</u>	<u>Value</u>
Mutual inductance to shield-VF coil (single turn)	M_{ds}	1.30 μH
Mutual inductance to trim coil (single turn)	M_{dt}	0.16 μH

C. Thermal Parameters

Conductor coolant	--	supercritical helium
Helium inlet temperature	T_{in}	4.5 K
Helium inlet pressure	P_{in}	5.7 atm

IV. Trim Coil

A. General Characteristics

1. Requirements

Life	t_L	10 years
Duty cycle	dc	87%
Max. overall ratio of magnet operating power to reactor thermal output	$P_{tc/o}$	10%

2. Other

Coil shape/type	--	superconducting coil located outside the TF coil, running in a direction parallel to the plasma
Outside radius of coils	R_{os}	12 m
Number of coils	N_t	16

B. Electromagnetic Parameters

Ampere turns (net)	A-T	0 MAT
Stored energy	W	16 MJ
Max. trim coil current (total)	I_c	6.4 MA
Current/coil	I_G	400 kA
Conductor material	--	NbTi in a Cu-CuNi mixed matrix
Self-inductance (single turn)	L_{tt}	3.02 μH
Mutual inductance to plasma (single turn)	M_{tp}	2.25 μH
Mutual inductance to shield-VF coil (single turn)	M_{ts}	3.06 μH
Mutual inductance to OH coil	M_{to}	0.11 μH
Mutual inductance to decoupling coil (single turn)	M_{td}	0.16 μH
Induced vertical field at plasma	B_{vt}	0.14 T

C. Thermal Parameters

Conductor coolant	--	supercritical helium
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	<u>Symbol</u>	<u>Value</u>
Helium inlet temperature	T_{in}	4.5 K
Helium inlet pressure	P_{in}	5.7 atm

3.2.5 Plasma Heating System

The function of the neutral beam injection system is to raise the plasma temperature to the ignition level. For the EPR, 50 MW of injected power into the plasma at 200-keV energy level is required for 5 sec each cycle. The reference design uses six beam lines, with two sources per line, injecting through six 40-cm-diameter apertures into the plasma. Each beam line includes a negative ion source, a neutralizer region, a bending magnet to deflect ions which are not neutralized, a vacuum system, and a drift tube to connect the beam line to the torus. No attempt is made to recover the energy of the deflected charged particle portion of the beam. By employing a negative ion source, which is yet to be developed, a beam line efficiency of 50% is projected. The back-up source is one based on positive ion technology currently in use with an overall efficiency of $\approx 15\%$.

Table 3.6 gives the plasma heating system parameters.

Table 3.6. Plasma heating system parameters

A. General Characteristics

1. Requirements

	<u>Symbol</u>	<u>Value</u>
Life	t_L	10 years
Duty cycle	dc	4.3%
Injected power, total	P_o	50 MW
Beam energy	E_b	200 keV

2. Other Features

Number of beam lines	N_L	6
Sources per beam line	S_L	2
Beam apertures into torus	B_A	~ 40 cm

B. Vacuum Parameters

Primary pump type	--	cryosorption
Gases pumped	--	hydrogen isotopes
Pumping speed, cryosorption pumps (total)	S_p	6,000,000 ℓ /sec
Cryosorption surface area	A	~ 120 m ²
Required vacuum	P_{vac}	$\sim 10^{-5}$ torr
Time between regenerations	t_{op}	24 hr

C. Electromagnetic Parameters

Injected power, total	P_o	50 MW
Beam energy	E_b	200 keV
Beam current (total)	I_o	250 A
Source type	--	negative ion

	<u>Symbol</u>	<u>Value</u>
Beam line efficiency	η_B	50%
Source current (total)	I_A	500 A
Source power (total)	P_A	100 MW
Pulse length	t_p	5 sec

D. Thermal Parameters

Cryosorption panel temp.	T_p	10-20 K
Chevron shield temp.	T_s	~ 77 K
Cryosorption pump charge time (cool-down time)	t_{CD}	~ 4 hr

3.2.6 Vacuum Pumping System

The function of the vacuum system is to pump the torus down initially and to maintain required vacuum conditions during operation. The system removes impurities and spent fuel (DT gas) at the end of each cycle. The roughing vacuum system is composed of Roots Blowers and mechanical pumps. The primary vacuum system is a cryosorption system with helium used to maintain the cryosorption panel at 4.2 K. Forty 1-m-diameter cryopanel are used with a total net pumping speed of 1,100,000 ℓ /sec.

The vacuum pumps as originally conceived are similar in design to existing commercially available cryosorption pumps, although our pumps are much larger. These pumps contain a molecular sieve coated panel operated at 4.2 K at the center. An array of chevrons which is cooled to approximately 77 K is placed near the pump entrance. A second array of chevrons at 15 K is located between the panel and the 77 K array to limit radiation heating of the sieve coating.

Table 3.7 gives the vacuum pumping system parameters.

Table 3.7. Vacuum pumping system parameters

A. General Characteristics

1. Requirements

	<u>Symbol</u>	<u>Value</u>
Life	t	10 years
Duty cycle	dc	100%
Availability - normal	--	98%
Plasma vacuum (operating)	P_{vac}	$\sim 10^{-3}$ torr
Torus vacuum (no injected gas)	T_{vac}	$\sim 10^{-9}$ torr
Gases pumped	--	hydrogen isotopes, helium, and impurities

2. Other Features

Primary pump type	--	cryosorption
Number of cryosorption pumps	--	40
Cryosorption pump diameter	d_p	1 m
Pumping speed, cryosorption pumps, total (net)	S_p	1,100,000 ℓ /sec

	<u>Symbol</u>	<u>Value</u>
Pump-down time (10^{-3} - 10^{-6} torr)	t_{pdp}	5 sec
Time between regenerations	t_{op}	24 hr
Roughing pump system	--	3 Roots Blowers + 3 mechanical pumps
Pumping speed, blower	S_B	5000 cfm
Pumping speed, mechanical pump	S_m	500 cfm
Pump-down time (ATM - 10^{-3} torr)	t_{pdr}	2 hr

B. Thermal Parameters

Cryosorption panel temp.	T_p	4.2 K
Chevron shield temp.	T_s	~ 77 K
Cryosorption pump charge time (cool-down time)	t_{CD}	~ 4 hr

3.3 Operating Chart Cycle

The EPR Operating Chart, Fig. 3.2, shows the sequence of events during a normal operating cycle, as well as the activities required to bring the EPR on line.

The normal operating cycle consists of injecting the fuel (DT gas), pulsing the ohmic heating coil to raise the plasma current, pulsing the neutral beams to raise the plasma temperature, the burn, pulsing the ohmic heating coils to extinguish the plasma current, and simultaneously evacuating the torus and charging the ohmic heating coil. During each portion of this cycle, the sequence chart indicates the performance of each major component, the electrical power required by the device, and the thermonuclear power to the steam cycle.

3.4 Power Conversion System

The EPR has a nominal thermonuclear power output of 410 MW(t) at a duty cycle of 87%, which is equivalent to an average power input of 356 MW(t), all of which was initially assumed to be transferred from the blanket to the power conversion system. Of the 356 MW(t) average power, 20% is removed in the first wall by cooling water entering at 38°C (100°F) and leaving at 121°C (250°F). The remaining 80% of the heat is removed by helium entering the blanket at an average temperature of 204°C (400°F) and leaving at an average temperature of 371°C (700°F).

An analysis based on these parameters indicates that utilization of the thermal energy from the first wall low-pressure water loop as input to a power conversion cycle would not be practical. Thus, this energy will be dissipated to the environment by evaporative cooling provided by mechanical draft towers. Practical utilization of the energy removed from the blanket by the water loop could be accomplished with increased temperatures in the primary coolant loops, as discussed in Sect. 3.6.

For 285 MW(t) of energy removed from the blanket by the helium loop, compressor power for circulating the helium is estimated to be 6.7% of the blanket thermal energy, or 19 MW. Thus, 304 MW(t) is available as heat input to the steam generator. An overall energy schematic pertaining to the thermal energy from the blanket is shown in Fig. 3.3.

The cyclic operation of the reactor will result in variations of the temperature of the helium

Fig. 3.2. EPR operating chart.

Fig. 3.3. Overall energy schematic for thermal energy from EPR blanket.

coolant. A thermal energy storage system is required to minimize variations in the rate of delivery of thermal energy to the power conversion cycle.

An intermediate heat transfer and energy storage loop would introduce an additional temperature difference between the heat source and the fluid in the steam generator, and would consequently result in a decreased thermal efficiency.

Another alternative considered in this study is a helium branch loop for energy storage, shown schematically in Fig. 3.4. This concept would reduce variations in the temperature of the helium available to the steam generator without introducing the penalty associated with an additional temperature difference.

The flow rates through the branch loop and through the parallel flow path will vary throughout the cycle, but the flow rate and temperature of the helium entering the steam generator will remain essentially constant. Thus, the storage capacity of the branch loop must be adequate to produce the necessary phase change in the temperature-time relationship for the helium entering the mixing point from the branch loop as compared with the temperature-time relationship of the helium arriving at the mixing point from the parallel flow path.

The length of time that the helium leaving the blanket will have a temperature lower than the 371°C (700°F) average temperature will be about 30 sec or more. This is due to the fact that the helium temperature begins decreasing gradually following the end of the burn portion of the cycle and begins gradually increasing again soon after the beginning of a new burn phase.

Over this period of time, flow-limiting devices will undergo gradual changes in settings to gradually increase the amount of flow through the branch loop. Correspondingly, they will decrease the flow through the parallel path to produce a constant flow rate of helium through the steam generator, while simultaneously changing the flow resistances in the parallel flow paths to maintain essentially a constant pressure drop from the exit of the blanket to the inlet of the steam generator.

The maximum flow rate required through the branch loop is estimated to be 87% of the total flow and the minimum is estimated to be 13% of the total. Thus, the flow-limiting devices will not be required to shut off the flow, and will therefore be less costly. The required storage volume in the branch loop is estimated to be 50% of the total helium flow for 30 sec, or about 934 m³.

The flow resistance through the branch loop may be made arbitrarily small by installing piping in parallel to reduce the velocity of the helium and simultaneously reduce the length of the flow path.

With the helium branch loop for energy storage, a power conversion cycle may be designed on the basis of a constant helium temperature of 371°C (700°F) at the inlet to the steam generator and a constant temperature of 193°C (380°F) leaving the steam generator and entering the compressor which circulates the helium.

3.5 Heat Balance

The steam power cycle selected (without the benefit of optimization) to operate with the helium loop specified above as a heat source has two turbine stages and one stage of reheat. The turbine cycle based on Ref. 4 is shown schematically in the heat balance diagram of Fig. 3.5.

A condenser pressure of 0.08 atm [2.5 in. Hg abs, at which the saturation temperature is 42.6°C (108.7°F)] was selected as a reasonable design value which might be used in conjunction with cooling towers. For a wet bulb temperature of 21.1°C (70°F), an approach to wet bulb of 5.6°C (10°F), and a range of 11.1°C (20°F) the condenser terminal temperature difference would be 4.8°C (8.7°F), a reasonable value.

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Fig. 3.4. Schematic diagram of helium coolant loop with branch loop for energy storage.

Fig. 3.5. Steam turbine cycle (schematic flow diagram and heat balance) for EPR.

Within the steam generator, the closest approach of the helium temperature to the temperature of the water in the turbine cycle is 4.6°C (8.3°F); this is occurring at the "pinch point" where the water is saturated liquid.

An external moisture separator is employed at the exit of the high pressure turbine, and the liquid water from the separator drains to the condenser. There are two stages of moisture separation in the low pressure turbine: one at 0.68 atm (10.0 psia), and one at 0.34 atm (5.0 psia).

The net thermal efficiency of the cycle is 28.4%, based on the generator output of 86.3 MW(e) and the thermal input to the steam generator of 304 MW(t). However, if the power input to the helium compressor of 19 MW(e) is deducted from the generator output, the remaining power output is only 67 MW(e), and the net efficiency [based on the average total blanket output of 356.5 MW(t)] is 18.8%. The net send-out and thermal efficiency are further reduced by power required for the condenser circulating water (cooling water) pump, cooling tower fans, and other station uses. The total of such uses may be estimated at about 4% of the generator output.

The steam cycle design parameters are tabulated in Table 3.8.

Table 3.8. EPR steam cycle design parameters

<u>Parameter</u>	<u>Value</u>
Average reactor power, MW(t)	356.5
Power to helium coolant, MW(t)	285
Power to pressurized water, MW(t)	71.5
Required helium compressor power, MW(e)	19
Power available to steam generator, MW(t)	304
Gross electrical power, MW(e)	86.3
Auxiliary power at 4% of gross, MW(e)	3.5
^a Net electrical power, MW(e)	63.8
Temperature of helium into reactor, °C (°F)	204.4 (400)
Temperature of helium from reactor, °C (°F)	371.1 (700)
Temperature of helium from steam generator, °C (°F)	193.3 (380)
Temperature of water into reactor, °C (°F)	37.8 (100)
Temperature of water from reactor, °C (°F)	121.1 (250)
Helium flow rate, kg/hr (lbm/hr)	1,184,000 (2,611,000)
Water coolant flow rate, kg/hr (lbm/hr)	736,000 (1,622,000)
Steam flow to turbine, kg/hr (lbm/hr)	395,000 (870,000)
Steam pressure at turbine inlet, atm (psia)	46.3 (680.8)
Steam temperature at turbine inlet, °C (°F)	260.0 (500)
Feedwater temperature, °C (°F)	42.8 (109)
Gross turbine cycle efficiency, %	28.39
^a Net plant efficiency based on blanket power, %	17.9
Number stages of feedwater heating	none

^aPower requirements for cryogenics, magnetic systems, shield cooling, etc., not included.

<u>Parameter</u>	<u>Value</u>
Turbine description	3600 rpm, reheat, 2 moisture separation stages, 23-in. last stage blade length
Generator description	3600 rpm, direct connected, rated at 0.9 power factor
Condensate flow at rated conditions, kg/hr (lbm/hr)	395,000 (870,000)
Cooling power	mechanical draft; evaporative
Approach to wet bulb, °C (°F)	5.6 (10)
Range, °C (°F)	11.1 (20)
Design wet bulb temperature, °C (°F)	21.1 (70)
Flow, kg/hr (lbm/hr)	1.7×10^7 (3.7×10^7)
Power for fans, kW (hp)	895 (1200)
Makeup rate, kg/hr (lbm/hr)	2.7×10^5 (6.0×10^5)

3.6. Alternate Power Conversion System

With a relaxation of the primary coolant maximum temperatures previously assumed and with the selection of a reasonable steam turbine cycle design, an improved thermal efficiency along with utilization of all of the thermal energy from the blanket as heat input in the power conversion cycle may be achieved.

The helium temperature increase through the blanket is again assumed to be 166.7°C (300°F) and the helium compressor work is assumed to result in an 11.1°C (20°F) increase in helium temperature. A boiler saturation temperature of 282.2°C (540°F) is assumed in an effort to make the cycle more closely resemble the typical cycles identified with fission reactors. Maximum steam temperatures of 740°F are somewhat arbitrarily assumed. A minimum of 11.1°C (20°F) is set as the difference between the temperature of the primary coolant fluids and the temperature of the water in the steam cycle.

The flow diagram for the steam turbine cycle is shown in the heat balance diagram of Fig. 3.6.

The net thermal efficiency of the steam cycle, based on heat input to the steam/feedwater at a rate of 375.5 MW(t), is 32.7%. With the helium compressor power of 19.0 MW(e) deducted from the generator output of 122.8 MW(e), the thermal efficiency is 29.1%, on the basis of 356.5 MW(t) heat output from the blanket. Again, net send-out will be reduced by station (in-house) uses of power for circulating pumps, etc.

The resulting temperatures for the helium coolant entering and leaving the blanket are 263.1°C (505.5°F) and 429.7°C (805.5°F), respectively. The calculated temperatures of the pressurized water entering and leaving the blanket are 53.9°C (129°F) and 207.2°C (405°F), respectively.

The parameters of the steam cycle proposed for modified coolant temperatures are given in Table 3.9.

The overall energy schematic, pertaining to the disposition of thermal energy from the blanket, is shown in Fig. 3.7. Note that the energy from the pressurized water loop used to cool the first wall of the blanket is now utilized.

Although this alternate power conversion system increases the net plant efficiency from 17.9% to 27.7%, a design evaluation has not been made for the impact of the high temperature coolants.

Fig. 3.6. Alternate steam turbine cycle (schematic flow diagram and heat balance) for EPR with modified coolant temperatures.

Table 3.9. Steam cycle design parameters for EPR with modified coolant temperatures

<u>Parameter</u>	<u>Value</u>
Average reactor power, MW(t)	356.5
Power to helium coolant, MW(t)	285
Power to pressurized water, MW(t)	71.5
Required helium compressor power, MW(e)	19
Power available to steam generator and preheater, MW(t)	375.5
Gross electrical power, MW(e)	122.8
Auxiliary power at 4% of gross, MW(e)	4.9
^a Net electrical power, MW(e)	98.9
Temperature of helium into reactor, °C (°F)	263.1 (505.5)
Temperature of helium from reactor, °C (°F)	429.7 (805.5)
Temperature of helium from steam generator, °C (°F)	251.9 (485.5)
Temperature of water into reactor, °C (°F)	53.9 (129)
Temperature of water from reactor, °C (°F)	207.2 (405)
Helium flow rate, kg/hr (lbm/hr)	1,184,000 (2,611,000)
Water coolant flow rate, kg/hr (lbm/hr)	400,000 (882,000)
Steam flow to turbine, kg/hr (lbm/hr)	396,000 (872,000)
Steam pressure at turbine inlet, atm (psia)	65.4 (961.5)
Steam temperature at turbine inlet, °C (°F)	393.3 (740)
Feedwater temperature, °C (°F)	43.0 (109.4)
Gross turbine cycle efficiency, %	32.70
^a Net plant efficiency based on blanket power, %	27.74
Number stages of feedwater heating	none
Turbine description	3600 rpm, reheat, 2 moisture separation stages, 23-in. last stage blade length
Generator description	3600 rpm, direct connected, rated at 0.9 power factor
Condensate flow at rated condicitions, kg/hr (lbm/hr)	396,000 (872,000)
Cooling tower	mechanical draft, evaporative
Approach to wet bulb, °C (°F)	5.6 (10)
Range, °C (°F)	11.1 (20)
Design wet bulb temperature, °C (°F)	21.1 (70)
Flow, kg/hr (lbm/hr)	2.0 X 10 ⁷ (4.3 X 10 ⁷)
Power for fans, kW (hp)	1040 (1390)
Makeup rate, kg/hr (lbm/hr)	3.2 X 10 ⁵ (7.0 X 10 ⁵)

^aPower requirements for cryogenics, magnetic systems, shield cooling, etc., not included.

Fig. 3.7. Overall energy schematic for thermal energy from blanket of EPR with modified coolant temperatures.

3.7 Facilities Description

A list of 20 reactor plant facilities was identified by reviewing the EPR reference design and design parameters. The facilities associated with the reactor and reactor systems were sized to reflect equipment concepts from the reference design. The balance of the facility was estimated considering overall plant operations. The conceptual arrangement of these facility blocks is shown in Fig. 3.8.

The main facility block, the reactor building, was given the shape presently used in nuclear power plants for such buildings. The reactor building will house the reactor vessel and related systems to produce a sustained nuclear fusion reaction. This reactor vessel and related systems will be contained within a biological shield. The primary consideration in sizing the building was the fact that the reactor must be disassembled for replacement of the blanket first wall. This disassembly will be performed remotely because of the radioactive environment.

For maintenance operations, reactor components will be transferred to the reactor maintenance building. To reduce the distance these reactor components will have to be moved, the reactor maintenance building was located adjacent to the reactor building. Reactor maintenance may entail replacement of the blanket first wall and other potential component modifications. The toroidal coil winding building has been located adjacent to the reactor maintenance building to utilize the handling systems during construction, and to give the capability of modifying the toroidal coil after the reactor is put in operation.

The Tritium Processing Building, Cryogenics Plant, Homopolar Generator Building, and Intermediate Bay contain equipment which services the reactor systems. The interdependence between the reactor and this equipment requires that these facility blocks be located adjacent to the reactor building.

The Steam Generator Building and Turbine Bay contain the equipment to utilize the heat generated from the nuclear fusion reaction for conventional electrical power generation. An additional item contained in the steam generator room is the helium branch loop piping for energy storage which provides constant temperature helium to the steam generator.

The balance of the facility blocks are required for a self-sufficient operation. A list of the contents and parameters used in sizing the facility blocks is given in Table 3.10.

Table 3.10. EPR facility contents

<u>Reactor building</u>	
Dimensions:	Diameter 184 ft Height 92 ft Hemispherical roof radius 92 ft
Contains:	Reactor (6 neutral beam injectors, cryosorption pumps for main vacuum system and neutral beam injectors, fuel injection system, torus, and magnetic confinement systems) Biological shield (above systems are contained within a hexagonal concrete biological shield) Polar bridge crane
<u>Reactor maintenance building</u>	
Contains:	Portable remote reactor assembly and disassembly tooling and equipment Reactor component transporters Blanket replacement tooling and equipment

Spare reactor component storage
Storage of scrap reactor components

Radwaste processing building

Contains: Gaseous radioactive waste processing equipment
Solid radioactive waste processing equipment
Liquid radioactive waste processing equipment

Intermediate bay

Contains: Compressors for helium (blanket primary heat transfer fluid) system
Heat exchanger between blanket cooling water and cooling tower
Circulating pumps for blanket cooling water
Heat exchanger between shield cooling water and cooling towers

Steam generator building

Contains: Steam generator
Heat exchanger between helium (blanket primary heat transfer fluid) and steam systems
Helium branch loop piping for energy storage
Gantry crane

Turbine building

Contains: Turbine
Specifications: 100 MW
single shaft
2 stages
double flow
23-in. last stage blades
3600 rpm
Generator
Specifications: 120 MVA
.9 Pf
direct coupling
Steam reheater and moisture separator
Condensor
Condensate pump
Lube oil systems
Gantry crane

Control room

Contains: Reactor controls
Plant controls
Cable spreading room
Safety-related ac and dc power supply

Tritium processing building

Contains: Tritium recovery systems
Tritium purifying system
Plasma fueling and injector system
Tritium storage system
Tritium waste processing

Cryogenics plant

Contains: Supercritical helium (4.5 K) systems
 Toroidal coil cooling system
 Primary ohmic heating coil cooling system
 Decoupling coil cooling system
 Trim coil cooling system
 Gaseous helium systems
 Main vacuum cooling systems
 Neutral beam injector cooling systems
 Liquid nitrogen systems
 Main vacuum cooling systems
 Neutral beam injector cooling systems

Homopolar generator building

Contains: Three homopolar generators (1 GJ each)
 Electrical switching equipment

Toroidal coil winding building

Contains: Toroidal coil winding machine
 Coil maintenance equipment

Research laboratories

Contains: Laboratories for evaluation of power reactor operation

Plant services building

Contains: Environmental heating, ventilating, and cooling equipment
 Plant offices
 Change house

Maintenance bay

Contains: Portable maintenance equipment for general plant maintenance

Shops & warehouse

Contains: Machine shop, welding shop, sheet metal shop, electrical shop, and refrigeration shop
 Warehouse for shop supplies, replacement parts, and general supplies

Diesel generator building

Contains: Two 250-kW emergency power diesel generators

Switch yard

Contains: Switch yard for electrical power produced by the Experimental Power Reactor

Supply switch yard

Contains: Switch yard for electrical power used by the Experimental Power Reactor

Cooling towers

Contains: Mechanical draft cooling towers

Cooling water pump house

Contains: Cooling tower circulating pumps

Fig. 3.8. Facility arrangement.

4. STRUCTURAL SUPPORT SYSTEM

In this section, we consider the structural mounting requirements of major reactor components. These structural elements do the following:

- (1) react to electromagnetically induced coil forces
- (2) support gravity loads
- (3) support inertial load induced by seismic excitation
- (4) provide an interface between systems to facilitate assembly
- (5) provide a mechanism for alignment control.

The mounting concept for each major component is listed in Table 4.1.

Table 4.1 Component mounting concepts

<u>Components</u>	<u>Mounting concept</u>
Blanket supports	Hanger rods attached to shield
Shield supports	Cantilevered posts attached to base of machine
Toroidal field coil supports In-plane forces	Cylindrical center support hub
Out-of-plane supports for torsional forces	Box beam and plates at top and bottom of TF coil which transfer torsional loads to center support hub
Poloidal coil supports	Built-up box beams for outer coil supports which transfer their forces to the TF coils and their supports
Machine-to-round supports	Posts between concrete foundation and machine to support gravitational and seismically induced forces

4.1 Structural Design Considerations

4.1.1 Design Criteria

Due to the conceptual nature of the work completed to date, only broad guidelines for structural criteria have been used. These guidelines were sufficiently complete to allow preliminary analysis considering major design loads including a fault condition of a TF coil. Before discussing the criteria used, various parts and functions of structural design criteria will be described. Structural design criteria consist of three major elements:

- (1) definition of all pertinent conditions, environments, and associated loadings
- (2) definition of appropriate safety factors, to ensure structural designs that provide acceptable risk to safety, environments, and economical operation
- (3) definition of analyses which are required to translate the environments and loads into appropriate static and time-dependent structural response parameters (stresses, strains, and displacements). This is done in order to relate them to material strength parameters or other design criteria (e.g., maximum displacements).

Three major functions are served by the structural design criteria:

- (1) They provide a "check list" to ensure that all pertinent loads and environments have been considered in the design and that they have been analyzed satisfactorily.
- (2) They establish appropriate safety factors or margins between the loads and material strength properties so that the risks associated with possible structural failure are commensurate with the consequences resulting from such a failure.
- (3) They provide a means for establishing a degree of uniformity between the many design organizations usually involved in a large project.

4.1.2 Mechanical Loading

Mechanical loading is defined here as any loading arising from the operation of the device, i.e., electromagnetic, pressure, and thermal. Only electromagnetic loads were used in this study, since they are the major loading on the support systems.

A sketch of typical coils showing current direction and the resulting electromagnetic forces is shown in Fig. 4.1. Figures 4.2, 4.3, and 4.4 give the intensity of the electromagnetic forces imposed on the coils and hence on the coil supports that were considered in this design.

The relatively small vertical field was found to have a major impact on the overall structural design. The TF coils which are optimized for the large in-plane radial forces have relatively little resistance to out-of-plane loading. The fault condition (zero current in a single TF coil) also resulted in substantial deformation and stress in the reference design structure.

4.1.3 Environmental Loading

Environmental loading is defined as those loads resulting from gravity, high winds, earthquakes, and other natural occurrences. With the exception of gravity, no environmental loading has been defined for an experimental fusion device; hence, only minimum consideration of these loads has been given to the EPR design. However, the capital investment and safety requirements will dictate that these effects be ultimately considered.

As with fission reactors, it is expected that primary emphasis will be placed on the seismically induced forces. In considering these forces, it must be recognized that designing a specific system is not a deterministic process but a probabilistic one. Seismic design requires a complex trade study involving variables such as (1) probabilities of various magnitudes of earthquakes occurring, (2) initial cost of the system with or without seismic design, (3) the damage cost of the system if not seismically designed against these probable earthquakes, (4) the risk to human welfare, and (5) life span of the system. The objective is to minimize the total loss including each of the above variables weighted according to its importance.

4.1.4 Codes and Standards

As with environmental loading, no specific code or standard has been identified as being applicable without modification to an experimental fusion device. General guidance may be obtained by referring to the Code of Federal Regulations as it applies to fission reactors and related apparatus. The EPR program will make necessary a code and standards effort for structural components (including their design, fabrication, and operation) similar to that which has proceeded in the fission reactor area. Work is under way to establish the character of such standards through projects like the Large Coil Program (LCP) of the Superconducting Magnet Development Program (SCMDP) at ORNL, and the Magnet Safety Studies at Brookhaven National Laboratory.

Fig. 4.1. EPR typical current and force directions.

Fig. 4.2. Distribution of radial force around TF coil perimeter (normal operation).

Fig. 4.3. Distribution of out-of-plane force on TF coil (vertical field of 1.0 kG).

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Fig. 4.4. Distribution of lateral load around TF coil perimeter resulting from adjacent coil with no current.

4.2 Design Description

There are two conditions which must be met by the supporting structure of a tokamak which make the design of this structure difficult. The first of these is that it must be maintained at cryogenic temperatures, requiring insulation to minimize the refrigeration load. The second requirement is that it withstand the enormous loads imposed by the various magnetic fields associated with the device. The largest of these loads is a constant centering force, 55 million lb for the present reference design, imposed by each of the TF coils. A more troublesome load results from pulsed operation of the poloidal field system. These fields impose loads on the toroidal field coils, requiring out-of-plane restraint.

The design of the EPR supporting structure requires a compromise between very close assembly tolerances, remote maintenance and assembly requirements, and thermal insulation. Undoubtedly there will be many changes in the design before a final version is reached, but the concept to be described here appears to be a workable design.

There are five parts of the major support structure which will be described. These five parts are (1) the center column, (2) the basic supporting platform, (3) the sockets around the column into which the superconducting toroidal field coils are nested, (4) the inter-coil space structure, and (5) the cryostats which must enclose every part of the structure (see Fig. 4.5). The entire cold structure communicates with the foundation (at room temperature) through column supports, each of which is enclosed in a cryostat to reduce the thermal losses by conduction through these columns.

Central Bucking Column. The primary function of the central column is to resist the centering forces of the 20 toroidal field coils. Basically, this column is a cylindrical structure approximately 12 cm thick and 4.5 m high with a 2.2-m inside radius. This column consists of two half cylinders of rolled stainless steel plate of standard dimensions. Each plate is 3 m long, 1.2 m wide, and 13 cm thick. The plates are drilled through the long dimension with cooling holes. The edge has a milled-out channel which acts as a coolant plenum for joining adjacent sections. These curved plates are welded together to form a half cylinder.

Along the two vertical edges of each half cylinder there is a key way. This key way is wider and deeper in one half cylinder than it is in the other, as shown in Fig. 4.6. When the two half cylinders are joined by a bolted key in these key ways, the larger key way provides a space which is filled with epoxy to form an electrical break in the cylinder.

Platform. The platform is a very simple structure made up of standard beam sections with plates completely covering the top and bottom of the beams. Under the location of each TF coil a 35-cm beam runs radially from under the center column to the outside circumference of the platform. Between these radial beams is a network of beams as chords of circles of different diameters. The beams have holes in the webs to allow passage between all volumes enclosed in the compartmentalized structure. These volumes are hermetically sealed by the plates so as to contain the coolant.

This platform presents a conducting shell which tends to compete with the plasma for the driving current. For this reason, an electric break, similar to the one in the center column, is included in the platform. The platform rests on pipe columns standing on the reactor foundation. The outer row of columns is equipped with lateral bearings on the top to permit movement of the platform with thermal changes.

Coil Sockets. The toroidal field coils bear on the platform for gravity support, and they bear on the center column for support of the constant magnetic load. In addition, the torsional loads imposed by the pulsing of the poloidal coils impose loads on both platform and center columns. To prevent this movement, these coils must be securely clamped or welded to the center column and the platform, and they must be interconnected wherever possible.

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Fig. 4.5. Elevation view of support structure.

Fig. 4.6. Support cylinder keyway.

The outside of the center column, as shown in Fig. 4.7, has a rather complex structure welded to it. Vertical members are welded all around the outside diameter of the center column. These members serve to establish the key ways into which the key on the front face of each TF coil inserts. Two curved plates for each coil weld to the vertical ribs on the column, and these plates provide the front bearing surface against which the TF coils center. On each edge of these two curved plates is a metal plate contoured to follow the coil shape and having two edges at right angles to each other. One of these edges is welded to the center column; the other is welded to the platform. These plates are in the general shape of right triangles, but with the hypotenuse curved to the contour of the TF coil. Between each of these side plates there are gusset plates to provide stiffness. These gusset plates have holes so that all compartments intercommunicate for cooling.

The clearance between the key on the TF coils and the key ways formed by the fabricated structure just described permits slipping the coils into their coil "sockets" without interference. Such clearances are unacceptable for pulsed operation of the coils. After the coils are in place, wedges are bolted between the noses of alternate coils forcing the keys against one side of the key way and locking the coils in place. These wedges are at discrete locations on the center column of the structure.

The cooling holes in the center column receive coolant by means of a communicating channel running all around the circumference.

Cryostat. A cryostat in the form of an evacuated space housing multilayered reflective insulation cannot be used around the TF coils where gravity or magnetic forces must be resisted. Also, it appears to be impractical to provide a common cryostat around all cryogenic components in the device. It was decided to provide a cryostat around the center column, one around the platform, an individual one around each TF coil, and a separate installation around the perimeter poloidal coil structures.

Almost half the circumference of each TF coil must have load-bearing insulation. At these locations alternate layers of Teflon and thin stainless steel reflecting sheets may be used. This has almost 400 times the thermal conduction of the best evacuated cryostat, but since the surface against which the coil rests is at 4 K, the heat loss is negligible. The non-load-bearing surfaces of the coils have conventional cryostat containment with the outer cryostat skin being continuous over the load-bearing insulation.

Coil Spacer Structure. The coil positioning and spacing is effected by the engagement of the coil key in the key slot of the center column. Since the key slot terminates at the top and bottom of the coil short of the vertical midplane of the coil, some additional inter-coil bracing is required. This bracing is accomplished by a box beam structure welded to the coil bobbins and the central column.

The box beam extends from each coil toward the beam from the adjacent coil. There is a gap between the beams to prevent interference in coil erection. When the coils are in place, this gap is bridged by welding a curved plate to each beam. To remove a coil, these plates must be removed by grinding out two structural welds.

Since these box beams present a large metal area connected to the coils, the beams must be enclosed in a dewar to prevent excessive thermal losses from the coil.

In addition to the main structure we have been describing, there is another structure which must operate at a cold temperature. This is the ribbed structure which contacts the TF coils on the outer machine circumference and which supports the external poloidal superconducting coils. These beams extend from the top to the bottom of the tokamak, and there is one over each of the 20 TF coils. Since these structures are in intimate contact with the 4-K TF coils and with the superconducting poloidal coils, the beams will have cryostats. It has not yet been decided whether or not to trace the beams with cooling lines.

- 1-BUCKING COLUMN
2,3-VERTICAL RIES
4-KEY CHANNEL
5-FACE PLATE FOR COIL
6-WEDGING CLAMP
7-COIL BOBBIN

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Fig. 4.7. Center column coil sockets.

The remainder of the structure is not maintained at low temperature and is quite conventional. It consists primarily of braced columns attached to the concrete foundation. On top of these supports are placed the shield sections. The blanket sections and the inner vacuum wall are supported off the shield.

4.3 Design Alternatives

There are three fundamental problems which must be faced in designing a satisfactory structure for the TF coils. The structure must support the coils against gravity, the very large centering forces, and the large out-of-plane forces resulting from the pulsed fields, faults, or other conditions such as earthquakes. In addition, access must be provided for large vacuum, injection, and miscellaneous other penetrations through the structure. Also, the inter-coil structure must be removable to permit replacement of internal components (first wall and blanket). A third restraint on the design concerns the heat leak to the 4.2-K coils and supporting structure. This extremely low temperature requires insulation to keep refrigeration power to a tolerable level. Several alternatives to the reference design were considered as follows:

1. Cable Support. An alternate TF coil support structure consists of a surrounding steel beam structure from which the TF coils are suspended by cables, possibly of graphite fiber construction. If the coils are hung in this way, it might be possible to allow the noses of the coils to nest, and to take the centering forces without the need for a center bucking column. This design might be inadequate to cope with a fault condition involving a loss of field in one coil.
The gravity load can be easily handled by the cable suspension method. It represents about the lowest thermal loss of any support scheme. Handling the out-of-plane loads by means of a system of restraining cables is the main problem. A cursory look at this indicates that the number of cables required is probably prohibitively large.
2. Two-point Support. It has been shown that with the proper shape of the TF coil it is theoretically possible to give it a two-point support to some type of central load column.⁵ This may require strengthening the coil between the supports on the inside circumference of the coil. This idea has been investigated very briefly. The attachment arms of the two-point support require such massive linkages that the problem of attaching them either to the coil or the column becomes very difficult.
3. Continuous Spacer Structures. This design concept would retain the bucking column and support platform, and would essentially fill the space between coils with solid plates either welded or bolted into place. These plates would have to be removed to replace the internal structures of the reactor. Since there are multiple large penetrations between each coil, large reinforcing rings would be required around these penetrations to provide adequate stiffness. This modification of the present reference design would probably take care of the out-of-plane loads, but would greatly complicate maintenance.

This method of restraint would also complicate the installation of dewars around the cold structure. Such a solid structure may indeed be required to immobilize the coils to a sufficient degree. If so, the implications to refrigeration loads and maintenance difficulties should be fully realized.

4. TF Coil Continuous Inner Support. In the reference design, a method of wedge clamps is described to force the key of the coil to one side of the key way for uniform support. Another way to do this is to cover the key as well as the flat coil face with a bladder. The coil is spaced away from the center column by some nominal clearance (0.25-0.50 in.). With

the coil in plane, the bladder is filled with an epoxy under some slight pressure. The TF coils, at room temperature, are then energized to some 10-20% of operating current. This magnetic force tends to conform the epoxy-filled bladder to all enclosing surfaces. When the epoxy sets, the coil is uniformly mated to the supporting surfaces.

4.4 Analytical Methods

As previously described, the EPR includes 20 oval-shaped TF coils evenly spaced about a toroidal axis. Each coil has associated with it a spacer structure as shown in Fig. 4.8, and adjacent coils are joined together by means of this structure. This assembly is mounted on a central hub structure, and is capped top and bottom by the secondary hub structure.

Due to the symmetry of the full assembly, it was possible to model only a quarter-section of the device and still perform an analysis of the full structure under the most general of loading conditions (within the linearly elastic region of the structure). The full finite element model developed for the study is shown in Figs. 4.9 and 4.10. The entire model was generated using plate and beam elements. The plate elements used were either triangular or quadrilateral bending and membrane plate elements. The beam elements were simple bending, torsion, and axial force members which include the effect of transverse shear.

The toroidal field coils were modeled using 36 beam elements of roughly the same length. Grid points along the coil length were chosen to conveniently match actual structure components either on the spacer or the hub. To account for the axial stiffness contribution of the superconductor, an effective area was defined based on the moduli of elasticity for the steel and conductor. Since the conductor is only half as stiff as the steel, its contribution to bending was ignored. This led to a conservative estimate for bending stiffness in the coil. Also, since interlaminar slippage effects could alter the bending contribution of the conductor in a relatively complex way, it was felt that such a conservative approach was appropriate. Though techniques do exist to handle frictional effects in stress analysis, no completely satisfactory method is available; thus the rationale for ignoring this contribution.

The coil-to-coil spacer structure was modeled using a simple layered grid work of plates and beams. Since the design calls for a box-beam structure which has an inner and an outer layer of surface plates, a hand calculation had to be performed to estimate the stiffness of the design structure. This stiffness was added to the beam finite elements which criss-cross along the edges of the plate in the finite element model. Despite the required hand calculation, the simplicity of the model, resulting from using only a single layer of grid points and elements instead of two layers for modeling the spacer structure, justified this approach.

The hub structure was modeled almost completely with plate elements as shown in Fig. 4.10. The use of plate elements in this section of the structure was fairly straightforward. The only beam elements used in the hub structure were sets of six beams (three top and three bottom) for each coil plane in the secondary hub. These were used to model the 3-in. plates along the key way directed normal to the coil as shown in Fig. 4.8.

Due to the relatively large ratio of hub thickness to coil cross-sectional thickness modifications were made to the hub plates. The aim of these modifications was to match the displacement/load behavior of the modeled hub to that of the design structure as closely as possible. This should improve the results of the analysis where the coil and spacer structure are concerned.

Reflective symmetry was used to analyze the symmetric substructure to get a complete picture of the stress state in the full structure. In brief, the method involves taking a general loading condition on a structure and splitting that loading into symmetric and antisymmetric components with respect to a given plane of symmetry. Once done, the loading subsets are applied to only the symmetric portion of the structure and the effect of the rest of the structure is accounted for by the application of

TOP VIEW

STRUCTURAL DETAIL

Fig. 4.8. TF coil structural system.

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Fig. 4.9. Finite element quarter model of EPR toroidal field coil structural system.

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Fig. 4.10. An exploded view of finite element model showing major components.

suitable boundary conditions along the plane(s) of symmetry. A detailed description of this technique has been documented in Ref. 6.

After the analysis was completed, a new capability was obtained by ORNL from NASA for the NASTRAN computer program. The new capability is called cyclic symmetry. The analysis described above required that one-fourth of the structure be modeled while the cyclic symmetry approach would have allowed modeling one-twentieth of the structure. The savings in both manpower and computer time are significant and will be utilized in future design analysis.

4.4.1 Analytical Results

The detailed structural analysis presented here represents an extension of the work reported in the EPR Scoping Study for FY 75. New areas considered this year were a fault condition and the effects of out-of-plane loads on the TF coils resulting from vertical fields.

A detailed analysis was not performed on the revised EPR structural support concept. For this report, a presentation of the results of a structural analysis of an earlier EPR reference design will be made for the purpose of documentation and critical examination. The earlier concept differs in having 24 TF coils, a 6.5-m major radius, and a 4-T field on axis. In addition, the poloidal field magnets are in a different arrangement in this earlier model than currently conceived.

The finite element model described in Sect. 4.4 and the loading conditions given in Subsect. 4.1.2 have been used to perform several structural analyses of the EPR toroidal field coil system.

The results presented are the outcome of five analysis cases run with the NASTRAN model previously described. These cases were chosen to allow evaluation of the effects of several loading conditions and model configurations. While by no means exhaustive, they nonetheless should provide useful information for future designs of the EPR toroidal field coil structural support system. The five cases are presented separately below with a description of what model configuration and loading was used. A discussion of these results and their relevance to future designs follows these five cases.

Case 1. The model used in Case 1 consisted of the toroidal field coils, the spacer structure, and the main hub, while the secondary hub structure was omitted. Supports were placed at the midplane of the coils. The compression hub thickness was set at 2 in. as called for in the initial design. Loading was due to the toroidal magnetic field under normal operation. Figures 4.11 and 4.12 show the induced stresses in the TF coils. Table 4.2 is a summary of the maximum stresses which occur in each of three major model sections: namely, the coils, the spacers, and the hub. These stresses are all considerably too large for the stainless steel structural material.

Case 2. The model configuration and support condition used in Case 2 was the same as that in Case 1, except that the thickness of the compression hub was increased from 2 in. to 5 in. The loading condition was also that due to the toroidal magnetic field only. The effect of this change can be seen by comparing the curves of Cases 1 and 2 in Figs. 4.11 and 4.12. The hoop stresses dropped a relatively small amount. On the other hand, the bending stress distribution changed considerably. Table 4.3 contains a summary of maximum stresses in the coils, spacers, and hub sections for Case 2. Comparison of Tables 4.2 and 4.3 shows the decided drop in maximum stress levels due to the increased hub thickness. While these stress levels are still high, the substantial reductions achieved seem to indicate that allowable stress levels for the in-plane normal operating condition loading could be reached with a moderate amount of design modification.

Case 3. The Case 3 model configuration is the full model of the TF coil structural system. It includes the secondary hub structure and the 2-in.-thick compression hub. The loading condition and support conditions were the same as those of Cases 1 and 2. As can be seen from a comparison of Cases 1 and 3 (Figs. 4.11 and 4.12 and Tables 4.2 and 4.4) no substantial effect on coil stress resulted from

Fig. 4.11. Hoop stress distribution in TF coil for Cases 1, 2, and 3.

Fig. 4.12. Bending stress distribution in TF coil for Cases 1, 2, and 3.

Table 4.2. Maximum stresses and displacements for Case 1

Maximum stresses			
Structure section	Stress type	Stress (psi)	
Coil	Hoop	37,500	
	In-plane bending	22,100	
	Out-of-plane bending	-----	
	Combined bending	-----	
	Total	56,000	
	Total at cavity edge	49,200	
Spacer ^a	Tension	10,300	
	Compression	48,400	
	Shear	27,600	
Hub ^a	Tension	127,000	
	Compression	249,000	
	Shear	128,000	
Maximum displacement (in.)			
Components (local coordinates)		Magnitude	
R	Θ	Z	
0.6	0.0	0.0	0.6

^aMembrane stresses only.

Table 4.3. Maximum stresses and displacements for Case 2

Maximum stresses		
Structure section	Stress type	Stress (psi)
Coil	Hoop	35,400
	In-plane bending	9,800
	Out-of-plane bending	-----
	Combined bending	-----
	Total	40,900
	Total at cavity edge	38,600
Spacer ^a	Tension	9,700
	Compression	36,400
	Shear	22,100
Hub ^a	Tension	88,700
	Compression	139,000
	Shear	114,000
Maximum displacement (in.)		
Components (local coordinates)		Magnitude
R	θ	Z
0.27	0.0	0.0
		0.27

^aMembrane stresses only.

Table 4.4. Maximum stresses and displacements for Case 3

Maximum stresses			
Structure section	Stress type	Stress (psi)	
Coil	Hoop	37,500	
	In-plane bending	22,400	
	Out-of-plane bending	-----	
	Combined bending	-----	
	Total	57,800	
	Total at cavity edge	50,900	
Spacer ^a	Tension	13,300	
	Compression	36,500	
	Shear	23,600	
Hub ^a	Tension	128,000	
	Compression	246,000	
	Shear	130,000	

Maximum displacement (in.)			
Components (local coordinates)			Magnitude
R	θ	Z	
0.59	0.0	0.0	0.59

^aMembrane stresses only.

the addition of the secondary hub structure. In fact, a slight increase in the maximum bending stress and total stress in the coils resulted.

The plots of in-plane displacements for Cases 1, 2, and 3 are shown in Fig. 4.13. Out-of-plane displacements were essentially zero due to symmetry.

Case 4. Case 4 was run using the same model configuration as Case 3. However, the supports for this case were switched to the base. The loading was changed to include the effects of the vertical magnetic field. As can be seen from the combined in-plane and out-of-plane bending stress plot in Fig. 4.14, the effect due to the relatively small vertical magnetic field is a disproportionately large value of stress at the outer edge of the spacer structure. Since there is little outer support structure to aid the TF coils in resisting the overturning torques induced by the vertical field, the main mechanism to resist these loads is the torsional and bending stiffness of the TF coils. This produces the large bending stresses shown. Since no hoop stresses are induced by the out-of-plane loading, the hoop stresses are the same as in Case 3. Also of concern is the large out-of-plane displacement of 4.11 in. shown in Fig. 4.15. As indicated by this plot, the coil takes a sharp bend at the edge of the spacer structure. It is in this region that the vertical magnetic field induces significantly high stresses (Table 4.5).

Case 5. Case 5 is the only case run with a fault condition loading. The fault condition assumed was zero current in a single TF coil. The results of this case are significant and this, along with other fault conditions, should be considered in future design studies.

The combined bending stress results of Fig. 4.16 show that the fault condition loading on the TF coils immediately adjacent to the faulted coil does indeed induce substantial out-of-plane bending stresses. As tabulated in Table 4.6, the induced stresses for this case are high.

A plan view of the deformation of the TF coils only under this fault condition loading is shown in Fig. 4.17. As can be seen, the coil begins bending sharply just after the edge of the spacer structure where the window begins. As in the previous case, it is in this region that the maximum stresses in the coil occur.

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Fig. 4.13. In-plane displacement of coil for Cases 1, 2, and 3.

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Fig. 4.14. Combined bending stress in coil for Case 4.

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(a) SIDE VIEW OF
DISPLACEMENTS

NOTE-PLANE OF COIL TIPPED 5°
ABOUT CENTER LINE FOR
EASE OF VIEWING

(b) PLAN VIEW OF DISPLACEMENTS

Fig. 4.15. Displacement of TF coil for Case 4.

Fig. 4.16. Hoop stress and combined bending stress in coil number 6 for Case 5.

Table 4.5. Maximum stresses and displacements for Case 4

Maximum stresses		
Structure section	Stress type	Stress (psi)
Coil	Hoop	37,600
	In-plane bending	22,100
	Out-of-plane bending	87,700
	Combined bending	94,800
	Total	121,000
	Total at cavity edge	77,100
Spacer ^a	Tension	26,600
	Compression	44,400
	Shear	28,300
Hub ^a	Tension	130,000
	Compression	251,000
	Shear	129,000

Maximum displacement (in.)			
Components (local coordinates)			Magnitude
R	θ	Z	
0.60	-0.07	-0.45	0.75
0.18	0.16	4.11	4.12

^aMembrane stresses only.

Table 4.6. Maximum stresses and displacements for Case 5

Maximum stresses			
Structure section	Stress type	Stress (psi)	Quadrant
Coil	Hoop	30,800	1
	In-plane bending	11,600	1
	Out-of-plane bending	57,000	1
	Combined bending	63,700	1
	Total	86,200	1
	Total at cavity edge	74,300	1
Spacer ^a	Tension	36,200	1
	Compression	43,900	4
	Shear	35,700	1
Hub ^a	Tension	11,300	1
	Compression	247,000	4
	Shear	115,000	4

Maximum displacement (in.)				
Components (local coordinates)			Magnitude	Quadrant
R	θ	Z		
-0.04	0.19	0.79	0.81	1

^aMembrane stresses only.

Fig. 4.17. Plan view showing displacements of coils under fault load condition.

Conclusions and Recommendations. A review of the analysis results as presented in the previous section leads to several conclusions and/or possible design recommendations. Some of these are discussed as follows.

First, the effect of the compression hub thickness on stresses is substantial. A hub stiffness which induces a zero radial component of displacement on the coil at some point within the region spanned by the spacer structure will probably be the optimum hub thickness as far as coil bending stresses are concerned.

For only in-plane loads on the coils, a constant tension, zero bending stress coil shape would be ideal for minimizing stresses provided the spacer structure as used in the reference design was entirely eliminated. Unfortunately, there are very substantial out-of-plane loads which the coils alone are unable to resist. This fact requires that extra support structure be added to handle these loads. As soon as these extra supports are added, the coil is effectively no longer an ideal shape. The whole concept of the ideally shaped coil is based on the assumption that the coil is free to expand uniformly, and the presence of an additional restraint at some point away from the central compression cylinder prevents this. This in turn causes bending stress. It is the presence of the spacer structure and, to a lesser extent, the secondary hub structure which dictates that an iterative process be used to minimize coil stresses. If the spacer structure were not present at all, then to a first approximation varying hub stiffness would merely change the rigid body displacement of the coil and would induce no additional stresses.

Next, some conclusions can be drawn regarding the spacer structure and secondary hub structure. The spacer structure of this design functions quite adequately in resisting a resultant out-of-plane loading such as produced by a non-conducting TF coil. The situation changes though for the overturning torques on the coils induced by the vertical magnetic field. Here the stiffness of the spacer structure simply causes the upper and lower sections to rotate almost as rigid bodies relative to each other, with the rotational stiffness between the two coming from the nearly vertical lengths of the coils spanning the window regions.

The implication here is that stiffer spacer structures would not significantly help the situation. However, three possible solutions to this design problem are apparent. These include (1) the addition of external torsional support structure (i.e., torsion about the toroidal axis), and/or (2) the strengthening of the TF coils, particularly in the area of the plasma/blanket access window, to handle the bending loads imposed by the vertical magnetic fields, or (3) a better connection between the spacer structure and the center hub. The first solution implies building additional superstructure, probably in the form of a torque frame such as is used on the Princeton Large Torus (PLT). The second solution might include designs in which the cross section of the coil is tailored to preferentially handle out-of-plane bending moments, in which additional steel is added to the outside of each coil casing where needed, and/or in which substantial shear reinforcement is provided across one or more of the windows.

The best solution might well be a suitable combination of each of the above techniques.

The relative rigidity of the spacer structure would allow attachment of an external torque frame at a small number of points on the structure. This would tend to simplify the type of frame needed and thus to minimize interference of the torque frame with window accessibility. Nonetheless, the addition of any external frame would tend to use up considerable amounts of space and would probably prove less practical than some other method of handling the overturning torque.

The fault condition loading considered in this study imposes a somewhat different design problem than that discussed above. In the fault condition case there is no net rotation about the toroidal axis which is inducing out-of-plane bending in the TF coils. In that case, the solution was to restrain

the rigid rotation of the upper spacer structure. With the fault condition the out-of-plane bending stresses come about from the direct action of the induced loads along the unsupported span of the TF coils across the window region. Since there is no way to predict which coil is going to fail, support must be added in the window region to every coil. It would probably be impractical to use an additional superstructure since the necessity of attaching the additional reinforcing to each coil at some point along the window would almost certainly interfere with window access. The same tailoring of the TF coil cross section that would help for the VF loading would obviously help here too. That, plus the addition of side ribs to the coil jacket, might be sufficient to bring the bending stress back down to allowable levels.

5. REMOTE MAINTENANCE AND ASSEMBLY

The EPR device will quickly become highly radioactive due to neutron reactions in materials; this activation dictates that remote maintenance methods be used for essentially all components. The practical feasibility of some remote maintenance tasks remains to be developed and demonstrated, because of the complex configuration, large size of individual components, need for accurate positioning and reliable joining, and limited space available for assembly/disassembly operations. Preliminary design studies have emphasized the great influence of the remote maintenance requirements on reactor design and the critical role of remote maintenance in determining plant availability and overall economics.

5.1 Outage Considerations

Remote maintenance methods are considered likely requirements for essentially all components integral to the device itself. Although considerable experience exists from the fission reactor program, the specific methods and equipment for remote maintenance of fusion devices remain to be identified. However, it has been recognized that remote maintenance will play a critical role in determining plant availability and will thereby influence the overall economics.

The sensitivity of electrical generation costs to the outage time assumed for blanket module replacement was investigated by extending the work of Kulcinski and Young.⁷ The Wisconsin study examined the effect of first wall lifetimes (of up to 8 yr) on the cost of electricity in UWMMAK-type fusion reactors, including the effects of removing and replacing blanket modules. An arbitrary assumption was made that an individual UWMMAK-1 module could be replaced in 3 days (or a total of 36 days to replace all 12 modules). An additional 5 days was assumed at each shutdown outage for cooldown and start-up.

To examine the sensitivity to outage time, the Wisconsin study was repeated assuming longer (but still arbitrary) module replacement times: module replacement times were increased first from 3 to 9 days per module, and then from 9 to 15 days per module. In each case, the time allowed for cooldown and start-up was maintained at 5 days. The evaluation was performed for assumed first wall lifetimes varying from 2 to 8 yr. The results from this effort provide a qualitative indication of the relative cost penalty associated with assumed blanket replacement times.

The cost estimates are based on the assumptions and methodology developed by the Wisconsin study, with one exception. The Wisconsin study determined the optimum outage time for each assumed first wall lifetime. For example, if the first wall lifetime is assumed to be 2 yr, then 12 modules can be replaced every 2 yr, or 6 modules every year, 3 every 6 months, 2 every 4 months, or 1 every 2 months. In extending the Wisconsin study, it was arbitrarily assumed that all 12 modules would be replaced at one outage, with that outage occurring at the end of the assumed first wall lifetime. Accordingly, the costs for the basic case examined in the Wisconsin study are somewhat higher, since their optimum outage replacement times were not used. However, all costs determined in the present study are on a consistent basis.

The cost estimates include the following four contributions:

- (1) operation and maintenance costs
- (2) fuel costs
- (3) return on capital
- (4) energy generated (kWh) per unit of time.

The operation and maintenance costs are increased by the need to replace the first wall; they therefore include the replacement material costs and the associated labor. The fuel costs are not directly related to the first wall lifetime. The capital costs increase because of (1) inventory cost of spare modules, (2) hot cell facilities to handle repair and/or replacement, and (3) long-term storage space

for the activated components. The energy generation costs are affected by the reactor availability associated with blanket module replacement.

The results of this study are presented graphically in Fig. 5.1. Case A presents information on the same basis as the original Wisconsin study, with the one exception previously noted. Cases B and C provide comparative information indicating the sensitivity of the cost of electricity to the time assumed for blanket module replacement. In developing these costs, the dominant factor driving the values upward is the lost revenue resulting from the outage time. The three other categories have a much smaller influence on the increased costs.

The results indicate that the differential in electricity cost is extremely sensitive to module replacement time for modest wall lifetimes (≤ 5 yr). These results reinforce the need for the development of long wall lifetimes (> 10 yr) and the need for incorporating remote maintenance considerations into all stages of the design development.

5.2 Design Impact

The entire mechanical design concept of the EPR has included the consideration of remote maintenance. Radiation damage is almost certain to make it necessary to replace the innermost part of the machine more than once during the operating lifetime. This requirement figured strongly in the basic design of the device.

The TF coils are very large, and have to be rigidly mounted to withstand the enormous forces imposed on them by the various magnetic fields to which they are subjected. These coils are enclosed in dewars, which are rather complex structures. For these reasons, the reactor was designed to permit the coils to remain in place unless the coil itself develops trouble. This requirement determined the size of the units making up the first wall and blanket, since these modules have to be small enough to move in and out between adjacent TF coils. The shield pieces are so rugged and see such a mild service that the probability of failure is quite remote. If one does have to be replaced a TF coil must be removed with it.

The blanket and first wall are assembled from 60 segments. This assembly involves clamping and welding by totally remote methods. Providing space in which to operate this remote equipment strongly influenced the detailed design of the blanket. The vacuum wall extends 4 in. on each side of the major blanket segments. The vacuum seal weld is made in this 20-cm gap between the blanket segments and then a filler blanket piece is put in place over the vacuum weld (see Figs. 5.2 and 5.3).

The shielding windings located on the inner surface of the shield have to be removed in order to get a blanket segment out. The need to do this affected the design of these coils and the mechanical framework on which they are mounted. In this removable location (between each TF coil) the shield coil segments are on a separate framework, and the shield is placed over the top of this structure.

The coil segments are made up of 2- x 2-in. copper extrusions, and have a 1/4-in. hole in the center for water cooling. Nine of these conductors are potted in a square to make up the conductors of the coil segments. These segments are bolted into the shield segments before the shield segments are installed in the reactor.

When the shield segments are put in place, each of the 16 coil segments extends beyond the edge of the shield on each side. The coil segments in one shield match the corresponding coil segments in the adjacent shield. The square bus has a special terminal fixture on each end of each segment. A special clamping clevis engages adjacent coil terminals, completing a mechanical and electrical joint between coil segments (see Fig. 5.4).

The shield, the TF coils, and most of the PF coils will probably last for the life of the machine. The exposed sections of the S-VF coils in the vacuum ports will possibly require replacement one or more times during the machine life.

Fig. 5.1. Effect of module replacement time on electricity costs.

Fig. 5.2. Vacuum vessel segments.

Fig. 5.3. Vacuum vessel seal.

Fig. 5.4. Shield winding mechanical joint.

It was first thought that the grinding of seal welds and rewelding these joints would pose one of the more time-consuming operations of replacing an inner wall. Further study has shown that these operations require a rather small percent (maybe 15-20%) of the downtime required for replacing these items. Much longer time is required to strip down the machine to get to these units. For example, in removing the neutral beam injectors there are a number of complex operations to be performed on each of the six injectors. For instance, approximately 40 water lines, 18 electrical connections, 14 refrigerated lines, and a roughing vacuum line must be broken. The unit must then be moved away from the torus.

There are numerous coolant lines, water, helium, and lithium exhaust lines which must be cut from each of the 60 blanket segments. The same is true for each of the 240 blanket filler pieces. The water cooling lines to the removable portions of the shield must be broken. The peripheral poloidal coils must be raised (those above the equator) or lowered (those below the equator) to clear space for blanket removal.

The vacuum ports (two between each coil) must be cut, a section removed, and an internal cut made to free the inner vacuum wall. Other diagnostic penetrations must be cut free from the vacuum wall. All of these operations must be done remotely, and will be very time-consuming. There is not sufficient experience in doing operations like this to be able to make reasonable predictions as to how long it will take. Of course, all the disassembly procedures must be repeated in reverse after the inner structures have been replaced.

With the present design, involving removal of sections between each TF coil, one of the time-consuming operations is the unclamping of 20 bolted connections on the S-VF coils at each of 20 locations around the torus.

An alternate concept to removing internals between each of the TF coils involves providing a means for rotating the blanket through 360°. This plan has not been fully developed, but a concept has been worked out, and from the study, it appears that this alternate scheme may be preferable. If this plan is adopted, access to the internals will have to be provided at only one of the 20 between-coil stations (see Fig. 5.5).

Another major design alternative still requiring further study is a secondary vacuum enclosure around the entire tokamak. Use of this approach would greatly simplify the remote welding required for high vacuum seals in the plasma vacuum chamber. In addition, the secondary vacuum vessel would eliminate the need for individual vacuum dewars around the TF coils and structural components, which poses a major design problem complicating remote maintenance. Questions of access for diagnostics, beams, repairs, etc. must be answered before a decision is made.

Design trade studies are recommended to provide a basis for the final design selection.

5.3 Assembly Sequence

In order to check the maintainability of the design, a step by step assembly sequence was developed. The sequence can be run backwards for the replacement of any component. In addition, a scale model was constructed to verify the assembly process (see Figs. 5.6 and 5.7). The major assembly steps are described below.

The building is complete with the usual cranes. The heavy concrete foundation with appropriate pipe chases and auxiliary cubicle space is in place. Also, necessary services (electricity, water, air, etc.) are piped to the assembly area. The site is ready for installation and assembly of the device.

1. Place on the foundation the superconducting (SC) poloidal coils that are beneath the machine platform. Place five SC poloidal coils on floor around outer circumference of machine.
2. Install support legs for cryogenic platform with their cryostats. Fabricate cryogenic platform with cryostat.

Fig. 5.5. Roller support for rotating blanket and shield.

Fig. 5.6. Scale model disassembly.

Fig. 5.7. Scale model assembly.

3. Attach coils that are beneath the structure to the cryogenic platform and fabricate bucking column and TF coil support structure complete with cryostat and helium cooling lines. Install shield support tripods with their leveling screw jacks.
4. Install solenoid, VF coils, and bucking coil windings in center of bucking column.
5. Erect TF coil on support jig (see Fig. 5.8). Move shield segment into coil and locate on support columns.
6. Move transporter to erection station. Transporter arms insert under top arc of shield. Removable gussets are inserted between top of shield and inner surface of TF coil. Arms are moved vertically, lifting shield off support legs, and lifting coil. The entire assembly is transported to the assembly area and lined up over a radial guide system. Locate and attach module to support stand. Clamp nose by tightening wedges, weld bottom of coil to support stand, and weld box beam at top between coils.
7. Repeat steps 5 and 6 for a second TF coil.
8. Clamp shielding coil joints on back 180° of shield (6 joints).
9. Install a vacuum wall segment in a blanket segment and locate inside the shield of TF coil #1, leaving seal weld of blanket wall exposed.
10. Bring up a second blanket and vacuum wall module and put in place adjacent to the first module.
11. Install remote welding jig in gap between blanket segments and make seal weld.
12. Install four quadrants of blanket filler in gap over seal weld and bolt to blanket frame.
13. Rotate the segment pair under TF coil #1.
14. Repeat steps 9 through 13 for TF coil #2.
15. Bring up a fifth blanket and vacuum wall segment and weld and clamp on both sides to modules already in place. This completes the vacuum wall and blanket under two TF coils.
16. Repeat steps 9 through 15 until the torus is completed.
17. Install the shield coil splices in their framework between the TF coils and make up clamps on each end of all splices.
18. Connect blanket coolant lines to cooling manifolds.
19. Weld all penetrations to vacuum wall and bring out sleeves through shield.
20. Install vertical ribs on lower half of torus and install five poloidal coils on these ribs.
21. Lower upper umbrella of PF coils.
22. Clamp poloidal structure to TF coils.
23. Complete electrical, cooling, and instrument connections, and all penetrations for pumps, etc.
24. Install injectors and other auxiliary equipment.
25. Locate all shadow shields around penetrations.

5.4 Program Recommendations for Research and Development

The review of the mechanical design of the EPR has led to a better understanding of the types of operations which will have to be performed in effecting remote maintenance. There are six categories of areas for which development programs must be inaugurated in order to solve the maintenance problems. These are: (1) transporting and locating large components, (2) remote welding and cutting of wide ranges of sizes and shapes, (3) mechanical clamping, connecting, and fastening, (4) viewing and inspecting, (5) instrumentation, and (6) waste disposal.

1. Transporting. The maintenance philosophy is that any part of the tokamak must be replaceable. This requires the use of transportation equipment capable of handling very large and very heavy components (Fig. 5.8). It is presently planned to accomplish this by surface transporters. There are devices that appear to be capable of being modified for this service

Fig. 5.8. Transporter and support fixture.

Because of the accuracy required for locating these items, it is likely that considerable design and development work will be required to adapt any existing transporters to this duty.

2. Welding and Cutting. There are many types of weldments that will have to be made on an EPR. Structural welds of stainless steel will be made in securing the TF coils. Probably the most critical welds are the circumferential welds on the vacuum vessel. There are 60 of these welds and each one must be vacuum-tight. They must be made in cramped quarters, and the development of an automatic welder to effect these seal welds is required. In addition, some kind of grinder, nibbler, or other parting tool must be developed to operate in the same space when it is necessary to replace the inner wall.

There are many coolant pipes that must be welded to coolant manifolds. These pipes range from about 1 in. to about 12 in. in diameter. Here again, the space is limited and very compact equipment will have to be developed to make the welds.

Another type of weld for which special equipment is necessary is the welding of sleeves for the numerous penetrations into the plasma. It is likely that the best process for these operations is a laser welder. The design is not yet firm enough to be sure that this type of welding will be required, but it appears to offer the best solution at this time.

3. Mechanical Fastening. In addition to weldments, there are many fastenings of a mechanical nature that must be made. The circumferential welds on the vacuum wall are just seal welds that will not provide any mechanical strength. The segments must be bolted together to provide this strength. A special jig for torquing these clamping bolts will have to be developed.

The shielding coils located on the inner surface of the shield are made up of short segments of water-cooled copper bus. These segments are joined by a connector between each shield segment. These connectors are forgings bolted to each coil segment. This bolting is critical. It must effect a good electrical joint and also have good mechanical integrity to withstand the forces from the magnetic fields.

These two mechanical joints (the vacuum wall and the shielding coils) are probably the most critical such joints in the reactor, but there will be very many mechanical connections of various kinds associated with connecting auxiliary services to the reactor. As the specific functions are defined, the required maintenance gear for mechanical fastening will be identified.

4. Viewing and Inspection. Every maintenance operation must be under the observation of maintenance personnel at all times. Where possible, three-dimensional vision should be provided. Special television equipment will be needed, and this must be packaged very compactly. It is probable that fiber optics will be required to allow observation of some of the most inaccessible places. Specific design of this equipment cannot be started until the design of the reactor is much more fixed, but the types of functions are evident now.

Observation of critical operations is a necessity, and appears to present no insurmountable problems. A more difficult task involves inspection and checking of a completed task. It is not at all obvious how the very numerous vacuum seal welds can be checked for quality. Methods as well as equipment must be developed for adequate inspection of maintenance operations.

5. Instrumentation. There will be many types of instrumentation required for the maintenance operation of a fusion reactor. It will be necessary to measure location and clearances with accuracy. The usual type radiation detection instruments will be required for safety. Instrumentation to monitor loads on maintenance equipment will be needed. Specific instruments cannot be developed until the design of the various jigs, fixtures, and manipulators is near completion.

6. Waste Disposal. Waste disposal presents a problem of considerable magnitude. When the inner wall is removed, it will be extremely radioactive. This is true for the vacuum wall and for much of the blanket as well. Both these items are bulky and must be cut up and compacted in order to occupy as little space as possible. These operations should not require extremely sophisticated machinery, but the capability for remote control must be included in the design.

6. PROJECT ENGINEERING

6.1 Preliminary Conceptual Engineering Plan

At the outset of FY 76, it was assumed that the EPR was to be constructed for operation in the mid-1980's. A preliminary engineering job plan was established to define the overall scope of the conceptual design in terms of manpower, cost, and schedule.

A detailed work breakdown was used as a basis for the cost estimate. In addition, a survey was conducted of some recent Oak Ridge area ERDA projects in an attempt to establish an additional basis for the overall cost of the conceptual design.

As a result of this effort, it was recommended that DMFE plan for an 18 month pre-conceptual design at a cost of \$3 million.⁸ This plan provided for the submission of a Schedule 44 Data Sheet and conceptual design report in September 1978 and start of the capital project in FY 80.

The elements of the pre-conceptual design activity are as follows:

- Establish a system for management and control of the project within the general framework of the RDT Standards.
- Develop a top level schedule.
- Establish overall project objectives and individual system objectives.
- Provide design criteria for each system and subsystem.
- Establish a reference design through an evaluation of system alternatives and trade-off studies.
- Identify R and D needs.
- Develop a working cost estimate and financial schedule.
- Provide an orderly manpower buildup for the conceptual design.

The required staff for the recommended program is shown in Fig. 6.1. A total cost of \$6 million is estimated for the conceptual design effort which would produce a Schedule 44 Data Sheet and a preliminary conceptual design report in September 1978.

A listing of program milestones is shown on Table 6.1 for the recommended funding level of the pre-conceptual design. Also shown is an additional funding (Level B) which assumes that the FY 76 funding is held at \$500,000, so that the majority of the pre-conceptual design is accomplished in an 18-month period starting in July 1976. The program milestones for the \$500,000 FY 76 funding, the level finally approved, are listed in Table 6.2.

6.2 Progressive Events Charts

These charts were prepared to outline the steps necessary to progress from the study phase through construction and final acceptance of an EPR. First, an overview chart, Fig. 6.2, is presented; this lists the progressive events generally required for all project systems. In this chart, it is the set of steps leading up to and including the Conceptual Design that is emphasized, rather than the specifics of implementation coming after Conceptual Design. Second, the Blanket and Shield System is outlined, Fig. 6.3, showing several specific events to illustrate a typical system.

One of the major values to be derived from such work is the focusing of efforts onto specific, system-by-system R and D program planning, including development, testing, qualification, etc.

Two major kinds of information are illustrated on these charts. One is the overall layout of device-independent steps such as "review reference design," "identify major components," "fabrication," and so on. The second kind of information is the system-specific information such as the particulars given on the Blanket and Shield Chart under "Provide Performance Characteristics to Interface Systems..." activities 35-36 and under "Initiate Experimental Program for Qualifying Blanket Design..." activities 50-52.

Fig. 6.1. Conceptual design manpower schedule.

These charts, although representing only the first of many steps, help to illustrate the major effort required to develop and implement a program plan for the EPR. Detailed logic diagrams followed by development of schedule-inclusive PERT-type charts are a necessary but costly step to maintain a large project on time and schedule. The study work this fiscal year continued to develop input for selected systems, but, due to limited funding, fell far short of the total effort required.

Table 6.1. Recommended EPR program milestones for pre-conceptual design period

<u>Item</u>	<u>Completion date</u>		<u>Remarks</u>
	<u>Funding level A^a</u>	<u>Funding level B^b</u>	
1. EPR project plan summary			
1.1 Project plan logic diagram	12/5/75	12/5/75	For each EPR system this will include an identification of the requirements for the design and qualification of the system, and an integration of all the systems into the EPR. In most instances steps required will be generic in definition and dates will be estimates.
1.2 Reference schedule	5/28/77	1/28/78	The best definition of effort required to place EPR into initial operation will be determined and included in this schedule. This will occur near the end of the pre-conceptual design.
2. Pre-conceptual design information			
2.1 Identification of EPR systems	10/31/75	10/31/75	This will be an identification of major functional pieces of EPR, such as TF coils, vacuum pumping, tritium processing, which will be used as the basis for developing design descriptions.
2.2 Identification of EPR subsystems	1/29/76	1/29/76	This will be an identification of the significant elements comprising each of the systems designated in Item 2.1.
2.3 Identification of components	2/26/77	12/3/77	This will be a compilation of major hardware items comprising EPR which will be useful in an early assessment of materials requirements, potential technology developments, schedule and cost estimates.
2.4 Identification of materials requirements	4/31/77	12/31/78	This will be an evaluation of the type and amount of materials required for EPR. This identification will permit assessment of the availability of basic resources, available production capability, etc.
2.5 Draft of system descriptions	6/4/77	2/11/78	This will consist of documentation by system of information similar to that identified in Sects. 1 and 2 of the SDD standards, supported by refinements in the current design.

<u>Item</u>	<u>Completion date</u>		<u>Remarks</u>
	<u>Funding level A^a</u>	<u>Funding level B^b</u>	
3. Assessment of technology			
3.1 Identification of major R & D requirements	12/26/75	12/26/75	This will be an identification of technology development required for EPR. It will be an ongoing process in which additional needs become apparent as the design evolves. The date has been selected recognizing the value of early identification based on existing information.
3.2 Technology program reallocation recommendations	4/3/76	4/3/76	This will be an identification of priorities in the CTR technology development program as indicated by EPR design requirements and probable Demo needs. These will be determined from refinements in the reference design and considerations developed for the Demo plant.
4. Cost assessments			
4.1 Assessment of component costs, including material requirements	6/4/77	2/4/78	This will follow from the completion of the items in Item 2.
4.2 Assessment of yearly operating costs for EPR	9/18/76	4/31/77	
4.3 Manpower and resource requirements for EPR	6/4/77	2/4/78	This will be an assessment of the staffing levels required as a function of time, for the duration of the project.
5. Design report	7/3/76 6/4/77	7/3/76 6/4/77 2/4/78	These reports will summarize refinements made on the reference design through advances occurring in the intervening periods. Specific parts of the design will be improved as needed; these will be incorporated into the periodic design updates.

^aLevel A - This assumes \$1 million spent in FY 76 (a decision effective 12/75) and \$2 million spent in the next 12 months.

^bLevel B - This assumes \$500,000 spent in FY 76, \$750,000 spent in the following six months, and \$2 million spent in the 12 months following that.

Table 6.2. EPR program milestones for FY 76, assuming funding at \$500,000

<u>Item</u>	<u>Planned date</u>	<u>Definition of milestone</u>
Reference design	9/9/75	Basic documentation of scoping study recommendations and point of departure for pre-conceptual design.
1. Project plan logic diagram	12/5/75	Identification of requirements for the design and qualification of the systems in EPR and the integration of all systems into EPR.
2. Identification of EPR systems	10/31/75	Identification of the major functional pieces of EPR which will be the basis for developing design descriptions.
3. Identification of EPR subsystems	1/29/76	Identification of the significant elements comprising each of the systems in Item 2.
4. Identification of major R and D requirements	12/26/75	Identification of the technology development required for EPR, based on existing information.
5. Technology program reallocation recommendations	4/3/76	Identification of priorities in the CTR technology development program, as indicated by EPR design requirements and likely Demo needs. These recommendations will be supported by needs developed through the refinement of the current EPR designs and by considerations established from current studies on the Demo plant.
6. Refine EPR design, and issue report	7/3/76	Investigations in the following areas: plasma engineering (plasma magnetism, start-up, fueling, etc.), blanket and shield (neutronics, effect of penetration, heat transfer, power conversion), remote handling (concepts for assembly, disassembly), and magnetism (conductor design, structural radiation damage, cryogenics).

ORNL EPR OVERVIEW SUMMARY PROGRESSIVE EVENT CHART

Fig. 6.2. ORNL EPR overview summary progressive event chart.

**ORNL EPR
BLANKET AND SHIELD
TYPICAL SYSTEM PROGRESSIVE EVENT CHART**

Fig. 6.3. ORNL EPR blanket and shield typical system progressive event chart.

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Appendix

Brief Description of the EPR Reference Design

M. Roberts (ed.)

BRIEF DESCRIPTION OF THE EPR REFERENCE DESIGN

As a result of intensive evaluation of the blanket and magnet systems of the September 1975 Reference Design, along with a continuing examination of the plasma engineering basis, an updated Reference Design has been generated. Although significant improvements have been made in the key systems, this updated design does not represent a full iteration of the design; in particular, there is no change in plasma shape or impurity control, both of which would be changed in a new Reference Design. A full description of the Reference Design is contained in ORNL/TM-5576.

1. Summary

The EPR is an ignited, D-T burning, power-producing, air core tokamak with superconducting toroidal field (TF) and, in large measure, poloidal field coils. The duty cycle is 87%, with a burn time of 100 sec. The machine has a major radius of 6.75 m, and a circular plasma radius of 2.25 m (aspect ratio of 3.0). A blanket is provided to transfer a nominal 410 MW(t) from the burning plasma to a steam power cycle. For design purposes, the nuclear aspects of the EPR are based on a neutron wall loading of $\sim 1 \text{ MW/m}^2$, which is equivalent to a power output of 800 MW(t). Operation at this power level is considered an optimistic upper limit. A neutral beam system is provided to inject 50 MW of power into the plasma to achieve ignition. Figures 1 and 2 illustrate the elevation and plan views, respectively. Figure 3 presents a composite chart of each of the key preparatory steps before operation and the key events during an operating cycle. An initial plant layout is shown in Fig. 4, identifying 20 facility areas necessary for the reactor and balance of plant.

2. Key Systems

2.1 Toroidal Field System

The toroidal field of the EPR is generated by 20 superconducting coils. These coils are arranged on a stainless steel structure to form

a torus, with the noses of the coils contacting a cylindrical center bucking column. The coils have an oval shape with a minor (horizontal) bore diameter of 7.4 m and a major (vertical) bore diameter of 9.7 m. These coils produce a maximum field at the winding of 11 T and an on-axis field of 5.0 T at the plasma. In the high field area the conducting elements are composite wires composed of superconducting Nb_3Sn filaments in a Cu-Sn matrix; in the lower field portion of the coil ($<6.5 \text{ T}$) superconducting NbTi filaments in a copper matrix are used. The conductor coolant is supercritical liquid helium used in a forced-flow mode.

2.2 Blanket and Shield System

The blanket and shield are located inside the toroidal field coils. The blanket is composed of 60 circular wedge-shaped segments which, when assembled, make up the torus. Each segment is further divided into 12 modules which constitute the basic building blocks of the blanket (see Fig. 5). The first wall of the blanket, which intercepts the non-nuclear energy, is cooled with low-pressure water. The liquid metal absorber portion of the blanket receives the high-energy neutrons from the plasma and is cooled by high-pressure helium.

The shield is located outside the blanket; it consists of 20 segments composed of stainless steel and lead. The shield is cooled by borated water and forms a thermal barrier between the high-temperature blanket and the cryogenic TF coils. Neutron flux coming through the blanket is attenuated by a factor of 2×10^3 . The shield segments serve as the mechanical support for the blanket, and these segments themselves rest on tripod support columns.

2.3 Poloidal Field System

The poloidal magnetic system is composed of the ohmic heating (OH) coil, the shield-vertical field (S-VF) coil, the decoupling coil, and the trim-vertical field (T-VF) coil. A major portion

Fig. 1. EPR elevation view.

① AIR CORE PRIMARY WINDING	⑤ T.F. COILS (CONDUCTOR, BOBBIN, DEWAR)
② DECOUPLING WINDINGS	⑥ SHIELD
③ V.F. TRIM WINDINGS	⑦ BLANKET
④ SHIELDING/V.F. WINDINGS	⑧ VACUUM CONTAINMENT

Fig. 2. EPR plan view.

Fig. 3. EPR operating chart.

of the ohmic heating and decoupling windings are located inside the center bucking column of the structure. The remainder of these windings, and all of the trim-vertical field windings, are located outside the TF coils, and their mounting structure connects to the TF coils.

The OH coil serves to generate the toroidal electromagnetic driving force on the plasma. With the expected plasma impedance, a flux change of 130 W in the central solenoid results in a 7.2-MA plasma current. The coil current is driven from fully positive to fully negative values. The S-VF coil delimits the magnetic fields from the plasma current, thus shielding the TF coils. The S-VF coil also produces the major portion of the vertical field maintaining plasma column equilibrium. The decoupling coil, closely coupled to the OH coil, is connected in series

opposition with the S-VF coil to cancel the electromagnetic force generated by the OH coil in the S-VF circuit. The T-VF coil provides an auxiliary vertical field supplementing the S-VF coil. Table 1 presents some key distinguishing features of these four related coil systems.

2.4 Beam, Vacuum, and Tritium Systems

The neutral beam injection system deposits 50 MW of power into the plasma at an energy level of 200 keV for 5 sec each cycle, to contribute to the start-up heating of the plasma and to raise the plasma temperature to the ignition level. Six beam lines, with two sources per line, penetrate the plasma through six 40-cm-diameter apertures. Each beam line includes a negative ion source, a neutralizer region, a

Fig. 4. EPR plant layout.

Table 1. Key features of the poloidal field system

<u>Coil</u>	<u>Conductor Type</u>	<u>Number of Windings</u>	<u>Location Key on Fig. 1</u>	<u>Sense Relative to Plasma Current</u>
OH	superconducting	1220	1	opposite
S-VF	copper	16	4	opposite
Decoupling	superconducting	16	2	same
T-VF	superconducting	16	3	not applicable (creates vertical field only)

Fig. 5. Schematic of blanket construction.

bending magnet to deflect ions which are not neutralized, a vacuum system, and a drift tube to connect the beam line to the torus.

The primary vacuum system penetrates the torus through two vacuum ports located between each TF coil; it is a cryosorption system in which helium is used to maintain the cryosorption panel at 4.2 K. The forty 1-m-diameter cryopanel provide a total net pumping speed of 1,100,000 liters/sec. The roughing vacuum system, which is used for initial pumpdown of the torus, consists of three 5000-cfm Roots Blowers coupled to three 500-cfm mechanical pumps.

A tritium handling system used to process approximately 2 kg of tritium per day is provided. The tritium handling system includes equipment to (1) evacuate the plasma vessel and the neutral beam injectors and maintain a vacuum there, (2) purify and recycle tritium and deuterium for the plasma fuel cycle, (3) recover tritium from experimental breeding modules, and (4) provide tritium containment and atmospheric cleanup.

3. Key Parameters

A tabulation of overall system parameters and their values for the EPR is shown in Table 2.

Table 2. Overall system parameters for the EPR

A. General Characteristics1. Requirements

	<u>Symbol</u>	<u>Value</u>
Device lifetime	t_L	10 years
Duty cycle	dc	87%
Availability - normal	--	80%
Thermonuclear power (ignition)	P_0	410 MW

2. Other Features

Plasma radius	a	2.25 m
Major radius	R_0	6.75 m
Aspect ratio	A	3.0
Plasma edge to winding distance	Δ	1.40 m
Plasma volume	V	675 m ³
Overall height	H	15 m
Overall diameter	D	23.3 m

B. Plasma Parameters

Burn time	t_B	100 sec
Current	I_p	7.2 MA
Confinement measure	$N_e \tau_E$	2.7×10^{14} sec cm ⁻³
Plasma temperature		
Ion temperature	T_{ion}	12.2 keV
Electron temperature	T_e	13 keV
Beta poloidal	β_p	1.9
Safety factor	q	2.5
Electron density	N_e	7.4×10^{13} cm ⁻³
Fuel ion density	N_i	7.0×10^{13} cm ⁻³
Impurity level	Z_{eff}	1.34
Injection power into plasma	P_b	50 MW
Injected deuteron energy	E_b	200 keV
Particle confinement time	τ_p	9.6 sec
Energy confinement time	τ_E	3.7 sec

C. Electromagnetic Parameters

Maximum toroidal field	B_{max}	11 T
Toroidal field on axis	B_T	4.8 T
Number of toroidal coils	N_C	20
Magnetic field ripple at plasma edge	ΔB	2.2%
Conductor configuration	--	composite cable in a square aluminum conduit
Superconductor		
in the high field region	--	Nb ₃ Sn in Cu-Sn matrix conductor
in the moderate and low field region	--	NbTi in Cu matrix conductor

Table 2. (cont.)

	<u>Symbol</u>	<u>Value</u>
Average poloidal field at plasma edge	B_p	0.64 T
Poloidal field coil core type	--	air
Poloidal coil volt-seconds	VS	165 Wb
D. <u>Thermal</u>		
Maximum power rating of blanket	P_R	800 MW
Blanket coolant	--	helium
Outlet temperature of blanket coolant	T_{out}	370°C
Blanket coolant pressure	P_{He}	70 atm
Shield coolant	--	borated water
Coil coolant (superconducting)	--	helium
E. <u>Nuclear Parameters</u>		
Neutron wall loading (operating point)	P_w	0.55 MW/m ²
Neutron wall loading (design point)	P_w	1.07 MW/m ²
Neutron flux on first wall (operating point)	ϕ_w	$2.4 \times 10^{13} \text{ n/cm}^2/\text{sec}$
Neutron flux on first wall (design point)	ϕ_w	$4.7 \times 10^{13} \text{ n/cm}^2/\text{sec}$
Tritium breeding ratio (in one experimental module)	BR	1.15

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