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TOKAMAK ENGINEERING TECHNOLOGY FACILITY SCOPING STUDY

by

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ABSTRACT

A scoping study for a Tokamak Engineering Technology Facility (TETF) is presented. The TETF is a tokamak with $R = 3$ m and $I_p = 1.4$ MA based on the counterstreaming-ion torus mode of operation. The primary purpose of TETF is to demonstrate fusion technologies for the Experimental Power Reactor (EPR), but it will also serve as an engineering and radiation test facility. TETF has several technological systems (e.g., superconducting toroidal-field coil, tritium fuel cycle, impurity control, first wall) that are prototypical of EPR.

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I. INTRODUCTION

An experimental power reactor (EPR) is the next major step beyond the tokamak fusion test reactor¹ (TFTR) in the current U.S. tokamak development plan² and is the focal point for the fusion technology development program. An EPR operating at or near net electrical power with an ~ 25 to 50% plant factor, as currently envisioned³⁻⁵, requires an extrapolation of plasma conditions (confinement, pulse length, etc.) well beyond those anticipated for devices that will be operational in the next few years (PLT, T-10, PDX, D-III, etc.) and extrapolations well beyond the current state of the art in certain essential areas of fusion reactor technology (superconducting magnets, energy storage and transfer, neutral injectors, impurity control, tritium control, first-wall, etc.). Such an EPR represents a significant national investment, probably costing well in excess of one-half billion dollars. To achieve operation of an EPR operating at or near net power with an ~ 25 to 50% plant factor, by the mid-1980's would be a bold stroke, which would go far towards enabling a demonstration of the commercial feasibility of fusion power to be made by the end of the century. However, such an undertaking would require significant increases in both the EPR design activity and in the supporting technology development activities and an acceleration of the upcoming tokamak experimental schedule, thus significant increase in the funding level of the U.S. fusion program. Moreover, major design decisions would have to be made before the bulk of the experimental results from TFTR are available, and several advanced technologies would have to be demonstrated in a fusion reactor environment for the first time on the EPR.

There would seem to be three general decisions that could be made with regard to the EPR, in order of decreasing ambitiousness: 1) a "net power" EPR could be built to operate in the mid-1980's, assuming that adequate funding can be obtained to support this undertaking; 2) an EPR with less stringent performance requirements, but which still embodied significant extrapolations towards reactor level in plasma conditions and technology, could be built to operate in the mid-1980's; and 3) the operation of a "net power" EPR could be delayed until the late 1980's so that TFTR results could contribute to the design basis, and a less expensive facility could be constructed in the early 1980's to demonstrate many essential technologies in a fusion reactor environment. The first of these options is being examined in

the conceptual design studies that are being performed at ANL, GA and ORNL during FY 1976. This report presents the results of a preliminary scoping study that was undertaken to identify the characteristics of a facility that could be built in the early 1980's in the third option.

The present study describes a relatively small tokamak reactor, called Tokamak Engineering Technology Facility (TETF), intended primarily to demonstrate fusion technologies required for the EPR. This device, with major radius ~ 3 m, utilizes an intensely beam-driven D-T plasma, and introduces, on a reactor scale, essential EPR technologies, i.e., superconducting magnets, first wall, impurity control, vacuum, closed-cycle tritium processing, complete remote handling, radiation shielding, and pulse lengths of several tens of seconds or longer. At the same time, the TETF would serve as a plasma and neutron radiation test facility, with a plasma flux comparable to that anticipated in EPR and higher powered reactors.

In order to produce the largest possible neutron flux with relatively modest injection energy (60 keV D⁰, 90 keV T⁰), in as small and inexpensive a device as possible, it is proposed to use the CIT (counterstreaming-ion torus) mode of operation for the plasma.^{6,7} In this operating regime, the D and T ions are found in nearly thermal distributions, oppositely displaced in velocity along the magnetic axis, and maintained by oppositely injected D⁰ and T⁰ neutral beams. A neutron flux in the range 0.5 to 1×10^{13} cm⁻²sec⁻¹ seems attainable in a 1.4 MA device, with modest electron energy confinement ($n_e \tau_E \sim 2 \times 10^{12}$ cm⁻³sec), and 30 MW of beam injection.

The principal plasma physics advance of the TETF would be long-pulse operation at high plasma temperature. Given steady-state beam injectors, quasi-stationary plasma operation is possible because the plasma is fueled entirely by the injected beams, and the counterstreaming ions, in principle, can be made to carry a large fraction of the net plasma current.^{6,7} In this respect, plasma studies in the TETF would be complementary to those in the TFTR.⁶ While CIT-type operation is not proposed here as an end in itself, it does make possible high-duty factor performance and provides significant neutron intensity in relatively small size. In fact, the proposed plasma size (current ≤ 1.4 MA) is characteristic of experimental machines presently coming into operation or under construction. For example, the performance of CIT-type plasmas with 40 keV injected beams will be tested (with D-D) in

forthcoming experiments in the PLT and PDX devices at PPPL. In the event that CIT-type operation is not achievable in practice, the TETF would resort to TCT-type operation with a performance penalty of 2-3.

The principal objectives of the TETF are: 1) to demonstrate EPR technologies in a fusion reactor environment; 2) to serve as a moderately intense 14 MeV neutron source that could be used to test small blanket modules and, over a long period, bulk-radiation damage; and 3) to serve as an intense plasma radiation source to investigate plasma-wall interactions. While the TETF would significantly advance many EPR-type technologies, it is intended that other technical features of the device, which are not extrapolatable to the EPR, be as close to the state of the art as practical. The electrical power requirement is sufficiently modest (~ 120 MW peak) so that substantial energy storage systems are unnecessary.

On the basis of the scoping study summarized in this report, it appears that the fabrication and operation of the TETF is feasible in the early- to mid-1980's. The technical feasibility of the TETF and its role in the U.S. fusion power program should be examined in greater detail.

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II. SUMMARY

The scoping study presented describes a relatively small tokamak reactor called the Tokamak Engineering Technology Facility (TETF). Principal geometrical parameters for TETF are given in Table II-1. The primary objective of the TETF is to demonstrate fusion technologies required for EPR, but it will also serve as an engineering and radiation test facility.

A. Plasma Physics and Reactor Performance

The TETF plasma will be fueled and heated by D and T neutral beams, injected parallel and antiparallel, respectively, to the toroidal magnetic field. These beams are used to stack large densities of oppositely traveling D and T ions, which together with neutralizing electrons, form a CIT (counterstreaming-ion torus) plasma. This system of counterstreaming ions has a large fusion reactivity, even for rather modest injection energies, because of nearly head-on nuclear collisions between D and T ions. This mode of operation offers the most powerful means of generating a large fusion-neutron flux in a small-sized tokamak with modest confinement parameters.

Ideal CIT operation is attainable only if the inward diffusion of low-energy neutrals and ions to the reacting region is small compared with the rate of energetic particle injection by the neutral beams. With an increasing population of low-energy ions, the reactor performance is degraded; in the limiting case, the plasma must operate in the TCT (two component torus) mode, fueled and heated by D and T injection. For injection energies ~ 60 keV, the ideal CIT has a Q (fusion power /injection power) about six times that of the TCT. The key to achieving optimal reactor performance is adequate pumping of the neutral gas in the scrape-off channel outside the discharge, in order to minimize the influx of recycled plasma. (Note that the ideal CIT system is sometimes denoted CBT for "colliding-beam torus.")

The present design specifies plasma operation in the transition regime between the ideal CIT and TCT modes. Injection of 15 MW of 60 keV D^0 beams and 15 MW of 90 keV T^0 beams maintains a plasma of $\langle n_e \rangle = 4 \times 10^{13} \text{ cm}^{-3}$ and $T_e = 4.5 \text{ keV}$ with energy confinement of $n\tau_E = 2.6 \times 10^{12} \text{ sec/cm}^3$. The Q value is 0.70, or 45% of the ideal CIT value. Other important plasma parameters are $B_t = 4.0 \text{ T}$, vertical elongation ≤ 1.5 and $I_p = 1.4 \text{ MA}$. The 14 MeV neutron production rate during the burn is $7.3 \times 10^{18} \text{ sec}^{-1}$, which leads to a peak neutron current of $7.1 \times 10^{12} \text{ n/cm}^2/\text{sec}$. Plasma performance parameters are summarized in Table II-2 for operation in the CIT mode.

TABLE II-1. Geometrical Parameters

Major radius, R_0	3.0 m
Plasma	
radius, a	0.55 m
elongation	≤ 1.5
scrape-off region	0.2 m
volume	30.7 m ³
Vacuum chamber	
nominal dimensions (oval)	1.5 m $\times$ 2.25 m
nominal wall area	131 m ²
test region on outside	~ 0.2 m $\times$ 0.8 m
dimensions with 0.2 m test region	1.7 m $\times$ 2.25 m
nominal volume	58.5 m ³
Magnet shield thickness	
inside	0.9 m
outside	1.05 m
top and bottom	> 1 m
Toroidal-field coils (12)	
bore (pure-tension "D")	3.65 m $\times$ 6.0 m
thickness (including cryostat)	0.55 m
Support cylinder thickness	0.17 m
Central core radius, r_v	
air core	0.48 m
including Cu OH coils	0.63 m

TABLE II-2. TETF Plasma Performance Parameters

Aspect ratio, $A = R_o/a$	5.45
Safety factor, $q(a)$	3.0
β_p (excluding alphas)	3.6
Magnetic field on axis, B_t	4.0 T
Plasma current, I_p	1.4 MA
Energy confinement, $\langle n_e \rangle \tau_E$	$2.6 \times 10^{12} \text{ sec/cm}^3$
Electron density, $\langle n_e \rangle$	$4.0 \times 10^{13} \text{ cm}^3$
Electron temperature, $\langle T_e \rangle$	4.5 keV
Q	0.7
Beam power	
60 keV D^0	15 MW
90 keV T^0	15 MW
Thermal power	51 MW
Power flow to wall	
14 MeV neutrons	0.130 MW/m ²
3.5 MeV alphas	0.017 MW/m ²
10-2000 eV charge-exchange neutrals	0.046 MW/m ²
10-100 eV electron heat conduction & particle flow	0.107 MW/m ²
10-100 eV D-T ion heat conduction and particle flow	0.076 MW/m ²
0.01 eV - 100 keV radiation	0.015 MW/m ²
14 MeV neutron current	
average	$5.7 \times 10^{12} \text{ n/cm}^2/\text{sec}$
peak	$7.1 \times 10^{12} \text{ n/cm}^2/\text{sec}$
Impurity control	low-Z coated first wall

All fueling is carried out by the injected beams, and the flux swing of the ohmic current transformer allows a burn time of ~ 1 min. with a duty cycle of $\sim 75\%$. The nominal burn cycle is given in Table II-3. Even longer burn times are possible when a significant portion of the plasma current is driven by the streaming ions.

The performance of the plasma specified in the reference design can be tested on forthcoming experiments in tokamak devices, now coming into operation or under construction, which utilize comparable-sized plasmas and powerful 40 keV neutral beam injection.

B. Reactor Description

The reactor, which weighs 2000 tons, is assembled inside the TETF facility. A cross section of the torus is shown in Figs. II-1 and II-2. The elliptical/oval-shaped plasma has a minor radius of 0.55 m and an elongation ratio of ≤ 1.5 . Surrounding the plasma is a type 316 stainless steel first-wall vacuum vessel consisting of twenty-four oval-shaped toroidal segments 2.2 m high by 1.7 m wide. These twenty-four segments are flanged and bolted together to form an irregular torus whose major diameter is 6 m. The inner surface of the vessel is coated with a low-Z material. An experimental volume, ~ 0.20 m wide by 0.8 m high, has been provided within the vessel in the outer wall region. First wall vessel segments have been designed to be removed and replaced with relative ease to extend the flexibility of the reactor as a test facility. All surfaces of the vessel are water cooled.

Shield blocks, ~ 1 m thick, surround the first wall vessel. The blocks are of a slab-type geometry. Ten pieces are used to shield two adjacent first wall segments for a total of 120 major blocks for the entire reactor. Inner and outer shield block arrangements are different. The inner blocks located between the first wall and the center core of the reactor are made of alternate layers of stainless steel and B_4C to give high radiation attenuation in the limited space. The outer shield blocks use lead mortar for bulk shielding for reasons of economy. The shield blocks are water cooled using a separate cooling system. Removal of the shield blocks has been simplified to accommodate experiments and first wall replacement. The outer shielding blocks can be modified to house many types of experiments with a minimum cost.

TABLE II-3. Burn Cycle Sequence

Inject gas and form target plasma at $I_p \sim 500$ kA	0-0.5 sec
Begin injection, raise I_p to 1400 kA, attain equilibrium plasma P	0.5-2 sec
Burn (continuous injection)	2-44 sec
Shut-off beams and ramp-down plasma current	44-46 sec
Exhaust torus to 10^{-6} torr and charge OH transformer primary	46-60 sec

Figure II-2. Schematic of TETF magnet design

Toroidal-field coils surround the reactor first wall and shield block assembly. Each coil is approximately 7 m high by 4.7 m wide and ~ 0.6 m thick, weighing ~ 35 tons. The TF coils are supported by an inner cryogenically-cooled cylindrical support spindle common to all TF coils and a lower support leg extending down and outward to the reactor building floor. The TF coil system includes load dump resistors, torque support frame, cryogenic and power supply systems.

The ohmic heating (OH) coil system consists of twelve ring coils and a central solenoid. All coils are water-cooled copper. The 6 m high central coil, weighing ~ 45 tons, is positioned in the center of the TF coil inner support spindle. Four thin ring coils are located above and four coils below the central coil. Two sets of larger ring coils circumscribe the reactor TF coils from above and below the midplane, the largest being 11 m in diameter. The outer OH coils are supported and restrained in the magnet support/torque frame, which is attached to the reactor floor. The inner coils are supported through the central pedestal foundation.

Twelve equilibrium-field (EF) coils complete the magnet coil systems. Eight coils are clustered adjacent to the inner OH coil sets; four above and four below the reactor, while four larger ring coils are distributed symmetrically around the outer extremities of the TF coils. The largest reactor coil is the outer EF coil whose diameter is 13 m and whose weight is 58 tons. The EF coils, like the OH coils, are of water-cooled copper design and are supported and restrained by the same magnet support/torque structure. Ancillary systems include the power supply, switch gear, power leads and water coolant system.

A large vacuum system is required to limit plasma recycling. The system that was studied and described below has known deficiencies, and alternate designs are being prepared. Twenty-four 0.75 m diameter ports are located symmetrically around the first wall between TF coils both above and below the reactor centerline. Each of the ports extends outward into an annular shield plug volume containing a quick-closing valve in its outer extremity. Trapped gases will flow through the annulus provided around the centered shield plug and out into a removable Y-shaped extension duct to two 25,000 liter/sec or larger cryosorption pumps. The pumps are operated one at a time allowing for on-line regeneration of the off-cycle unit. Isolation valves are provided to allow for this regeneration purpose. In addition to the cryosorption pumps, a

series of blowers and diffusion pumps are supplied for initiating the pump-down in the first-wall torus from atmospheric conditions. The design provides a method to accumulate fuel from all components including any leakage of fuel from the vacuum system during any of the operational procedures. Maintenance and repair of all vacuum components is accomplished through isolation, removal and replacement using special remote-operated flange connections, valves and lifting dollies.

Eight neutral beam injector units, each containing two ion sources, ring the reactor. The beams are aimed tangentially into the plasma in a symmetrical pattern of clockwise and counterclockwise direction. Each beam enters on the centerline of the vacuum vessel at an angle of $\sim 45^\circ$ to the line of tangency through an 0.55 m wide by 0.8 m high rectangular opening, grazing the inside plasma toroidal centerline. The injector duct is shielded as it leaves the reactor. Each injector is attached to the reactor with an isolation valve. Each beam line consists of a source, accelerator, separator, neutralizer and ancillary vacuum and energy recovery equipment. Two lines are stacked in each injector and they aim into the torus from 6° above and 6° below the median plane. The beam length to the reactor first wall is ~ 7 m. Major power supply equipment is located in the adjacent building. However, hookup is made with special lead lines and step-up transformer units placed at the base of the injector units. A neutral-beam test facility is located on the reactor floor within the reactor containment shell. This area, enclosed by radiation shielding, contains remote maintenance and hookup equipment and a permanent power supply.

C. Magnet Systems

The TETF magnet system consists of three separate sets of coils; the superconducting toroidal-field (TF) coils, the ohmic-heating (OH) coils and the equilibrium-field (EF) coils.

Superconducting TF coils are required to avoid huge power consumption and large capital cost in power supply. NbTi is the most suitable superconductor because of its ductility. The maximum goal of the TF coil system is to provide a peak field of 10 T (4.0 T at plasma) operating at 3°K . A novel 3°K refrigeration system using supercritical dense helium gas as the working fluid is proposed. The minimum goal of the TF system is to provide a peak

field of 8 T (3.25 T at plasma) operating at 4.2°K. Characteristics of the TF coil system are summarized in Table II-4.

Since it is not clear that ac superconducting coil technology will be developed in time for the TETF fabrication, a water-cooled copper coil design is chosen for the EF and OH coils, and a superconducting coil system design is included as a design option to reduce the power demand. The water-cooled copper EF coils are placed outside the TF coil so that coil assembly and disassembly, blanket and shield assembly and disassembly, and first wall assembly and disassembly will be much easier. The EF coils are decoupled magnetically from the OH coils and provide 7.8 volt-sec to the plasma, thus reducing the volt-sec requirement for the OH coils. The water-cooled, copper OH coils are located outside the TF coils and arranged to impose a minimum field and torque on the TF coils. The total volt-sec required during the startup is 12, of which 4.2 volt-sec is supplied by the OH coil. During the burn cycle, the OH coil delivers 5 volt-sec to maintain the plasma current over ~ 1 min. burn period. Characteristics of the EF and OH coils are summarized in Table II-5. This design allows the total power grid capacity of the TETF to be about 100 MW, so that no major cost in the energy storage and power supply is incurred.

D. Neutral Beam Injectors

The neutral-beam injector system for TETF consists of eight injectors with two sources per injector. Four of the injectors (eight sources) provide 15 MW of 60 keV D^0 beam; the other four injectors provide 15 MW of 90 keV T^0 beam. Each ion source is followed by a double focusing bending magnet to separate molecular ions from atomic ions, a neutralizer for the atomic ions and a second bending magnet to remove ions from the beam emerging from the neutralizer. Direct energy converters are used to recover energy from the ions removed by the first and second separators. The two neutral beams from each injector converge toward an 0.55 m wide, 0.8 m high aperture in the first wall of the torus at angles of 6° above and 6° below the horizontal median plane of the torus. Characteristics of the neutral beam injector system are summarized in Table II-6.

TABLE II-4. TETF Superconducting Toroidal-field Coil Parameters

Conductor	
design	graded
type	braided cable
stress in stainless steel support	45,000 psi
stabilization	cryostatic
Materials	
superconductor	NbTi
stabilizer	Cu
insulator	fiberglass epoxy
support	316 stainless steel
Coil dimension	
bore (pure-tension "D")	3.65 m × 6.0 m
thickness (including cryostat)	0.55 m
Number of coils	12
Field ripple, maximum	0.99%
Access between coils, maximum	2 m
Operating temperature	3°K/4.2°K
Peak field	10 T/8 T
Superimposed ac field	0.2 - 0.4 T
Operational current (10 T)	10,000 A
Average current density (10 T)	2650 A/cm ²
Ampere-turns	
total	59.5 MA-turns
per coil	4.96 MA-turns
Stored energy	
total	3100 MJ
per coil	258 MJ
Weight per coil	35 tons
Coil winding	helical layered
Coil protection	series

TABLE II-5. TETF Water-cooled Copper Poloidal-field Coil Parameters

Equilibrium-field Coils

Conductor	
material	Cu
volume	20.4 m ³
current density	1500 - 750 A/cm ²
Power dissipation	
instantaneous	21.5 MW
cycle-averaged	15 MW
Inductive stored energy	68 MJ
Operational current	30 kA
Operational voltage	5 kV
Ampere-meters	153 × 10 ⁶
Volt-sec to plasma	7.8 volt-sec
Field at plasma	.28 T
Number of coils	12

Ohmic-heating Coils

Conductor	
material	Cu
volume	6.3 m ³
current density	2534 A/cm ²
Maximum peak field	6.37 T
Power dissipation	
beginning of cycle	54 MW
end of cycle	77 MW
Inductive stored energy	116 MJ
Operational current	30 kA
Operational voltage	5 kV
Ampere-meters	321 × 10 ⁶
Volt-sec to plasma	
startup	4.2 volt-sec
burn	5.0 volt-sec
Number of coils	12 plus solenoid

TABLE II-6. TETF Neutral Beam Injection System Parameters

	<u>D⁰</u>	<u>T⁰</u>
Neutral beam power (MW)	15	15
Neutral beam energy (keV)	60	90
Number of injectors	4	4
Number of ion sources per injector	2	2
Ion beam per source (A)	69	46
Beam composition ($D^+/D_2^+/D_3^+$)	0.75/0.18/0.07	0.75/0.18/0.07
Plasma emission current density (A/cm ²)	0.300	0.245
Type of grid	multi-aperture	multi-aperture
Tritium to torus (gm/hr)	-	~ 15
Gas efficiency	0.45	0.45
Pulse length	dc	dc
Overall neutral injector efficiency	0.56 - 0.59	0.56 - 0.59

Each beam enters the toroidal vessel through a shielded, water-cooled duct after passing between two TF coils. Any neutral beam line can be isolated and removed without contaminating the toroidal atmosphere. The malfunctioning beam injector can be transported to the hot cell for repair and then can be tested in the neutral beam test facility without disturbing the TETF operation.

E. Power Supplies

The power supplies for the OH coils and EF coils are based on normal conducting copper windings with an optional superconducting design. An important feature is that the level of powers for both coils are within the range of operation for pulsing directly off the power line.

The OH coil provides 4.2 volt-sec to heat the plasma and 5.0 volt-sec to maintain plasma current during the burn phase, and is charged to 30,000 A in one direction prior to heating and discharged to zero in two seconds for heating. After crossing zero field, the coil current is increased slowly to maintain the plasma current. A full 5.0 volt-sec requirement during burn requires a coil current of 35,700 A. Radiative power loss in the coil reaches 53.9 MW at 30,000 A. The coil time constant is 3.35 sec.

The power supply envisioned to meet these requirements is a phase-controlled, polyphase SCR rectifier driven directly off the power line. The main constituents of the power supply are the polyphase transformer, the SCR rectifier banks, the SCR reversing switches and an optional filter. The power supply performs in three operating modes: preheating, coil-charging mode; the heating, rapid discharge mode; and the burn phase, slow-charge mode. In the preheating, coil-charging mode, the rectifier phase angles are controlled to give the maximum positive charging voltage and minimum charging time for the coil. The peak voltage for 5.0 volt-sec is 2150 volts and the magnet takes about 6 sec. to charge to 30,000 A at 2150 volts.

For rapid discharge, the firing angles on the rectifier bank are altered to turn on with negative voltage on transformer terminal. This directs power back into the power line from the magnet. This mode continues as long as current flows in the positive direction through the rectifiers. When the current crosses zero, the reversing SCR's are activated and the phase angles on the main SCR rectifiers are returned to forward bias so that slow

charge of the OH coil for the burn cycle can begin. At the end of the burn cycle, the rectifier bank is again phased back to return magnet energy to the line.

The EF coil is designed to operate at 30,000 A and store 68 MJ of energy. The peak resistive power loss in the coil is 21.5 MW, but because the magnet must be able to reach 30,000 A in 2 sec., the possibility exists for drawing as much as 79.3 MW from the line. However, the time for which EF coil power is increasing roughly corresponds to the time that the OH coil is feeding energy back into the line. The system is designed to take advantage of the energy balance to reduce the peak power demand during this 2 sec. period.

After the EF coil reaches full current, the coil voltage can be reduced to just satisfy resistive power requirements. Again, the power supply would be a phase-controlled, SCR rectifier type without the reversing circuits. The peak charging voltage required by the EF coil power supply is 2650 volts.

Superconducting OH/EF coils are being considered as an optional design. The operating currents and stored energies remain the same for the two coils.

One power supply option for superconducting coils would operate off the power grid with the same type of system required for normal coils. Since no resistive losses occur with superconducting coils, the power taken from the grid to the OH coil during magnet charging is smaller than with a normal conducting coil. However, power returned to the grid during the ohmic-heating period is much larger because coil energy is not consumed by resistive heating. As a consequence, the peak capacity of the ohmic heating supply must be larger; roughly, 107 MVA as opposed to 77 MVA for the normal conducting coil. Peak megavolt amperes of the supply can be considerably reduced by switching on a resistor to assist in OH coil discharge during the ohmic heating period. A 0.16 Ω resistor to assist the OH coil discharge when switched into the circuit would reduce the peak power supply capacity from 107 MVA to 43 MVA.

The EF coil power supply peak capacity would be somewhat smaller with superconducting coils because of the lack of resistive power losses. The required capacity of the power supply for superconducting EF coils is about 67.5 MVA as compared to 80 MVA for normal conducting coils.

The TF coil power supply represents no major design difficulty. Its size is entirely dependent on how rapidly the coil is to be charged. A 70 kW supply will charge the coil in twenty-four hours and the 425 kW supply will charge the coil in four hours.

The neutral-beam power supply must provide approximately 22.5 MW at 60 kV for the D beams and 22.5 MW at 90 kV for the T beams. The power is taken off the electrical grid. Polyphase SCR rectifiers are the most likely choice for developing these voltages. Spark protection probably can be provided by vacuum switch tubes for these voltage ranges. A high frequency polyphase SCR rectifier with a saturated time delay transformer (STDT) could be considered as an alternative to the above system.

Characteristics of these power supplies are summarized in Table II-7.

F. Neutral Injector Vacuum System

Pumping is required for each neutral beam injector (2 beam lines per injector) on the inlet side and on the exit side of the neutralizers. The D_2 gas load on the inlet side is 45.8 Torr l/sec. An average pressure of about 2×10^{-5} Torr is adequate so the required pumping speed is 2×10^6 l/sec per injector. This rate of pumping speed requires 40 m² of cryosorption panel. At the neutralizer exit, the D_2 gas load is 10.2 Torr l/sec. With an average pressure of 1×10^{-5} Torr, the pumping speed required is 1.2×10^6 l/sec per injector. Twenty-four (24) m² of cryosorption panel will provide this pumping speed. Characteristics of the neutral beam vacuum system are given in Table II-8.

Impedance of the gas flow to the panels is reduced by placing the panels within the neutral beam enclosure itself on the vertical side walls of this enclosure. The pumping surface is divided equally between both sides of the enclosure to ensure more uniform pumping.

Regeneration of the cryosorption panels will be necessary periodically. To avoid interruption of reactor operation at this time, cryosorption panel area is doubled and the means provided for isolating half of the area for regeneration while the other half is pumping.

G. Toroidal Vacuum System

The reactor vacuum system pumping arrangement contains twenty-four ports of 0.75 m diameter each, positioned vertically between magnets, both above and below the vacuum vessel, to provide access for pumping. Each of the

TABLE II-7. TETF Power Supply Parameters

Power pulsed directly off-line		
power		100 MW
voltage		132 kV
OH coil system (copper coils)		
type		polyphase SCR rectifier
peak voltage		2150 volt
EF coil system (copper coils)		
type		polyphase SCR rectifier
peak voltage		2650 volt
TF coil system		70 kW
Neutral beam injection system accelerators		
type		polyphase SCR rectifier
D injection (60 kV)		22.5 MW
T injection (90 kV)		22.5 MW

TABLE II-8. Neutral Beam Vacuum System Parameters

	<u>D⁰</u>	<u>T⁰</u>
Number of injectors	4	4
Inlet side of neutralizer		
gas load/injector (Torr l/sec)	46	30
pumping speed/injector (l/sec)	2×10^6	1.33×10^6
cryosorption panel/injector (m ²)	40	26.7
Exit side of neutralizer		
gas load/injector (Torr l/sec)	12	7.2
pumping speed/injector (l/sec)	1.2×10^6	7.2×10^5
cryosorption panel/injector (m ²)	24	14.4

ports is fitted with a plug valve, extension Y-type duct, and a set of two cryosorption pumps--each with an isolation valve. Only one pump at a time is in service, while the other is being regenerated. Ancillary systems include the valve actuator power supply, coolant lead lines and seals, annular shield plug support, duct shielding, cryogenic supply leads, supplementary roughing pumps and blower connections, regeneration heaters, and system lines for containment and recovery of fuel.

All major vacuum equipment will be located within the reactor containment shell, with the exception of the helium compressors. This is a safety measure to minimize hazards.

Optimal performance of the plasma requires a neutral density in the scrape-off channel that will not exceed 3×10^{11} atoms per cubic centimeter. This density corresponds to a pressure of 1×10^{-5} Torr at room temperature. Such a pressure with a gas load of 60 Torr l/sec requires a pumping speed of 6×10^6 l/sec--a speed much greater than can be expected from the reference system (see Table II-9).

With appropriate pumping surfaces, it would be possible to maintain the required density--provided that the means is made available above the aperture to de-energize fast neutrals so that they can then be pumped. At least 20 m² of cryosorption panel would be necessary above each aperture.

If the power required to maintain a 4°K temperature on the cryosorption panels in the face of high neutron flux is prohibitive, it may be possible to use gettering surfaces such as titanium or zirconium to accomplish the pumping. The energy entering the gettering surfaces could be used to maintain the surface temperature at the level required to activate the getters for efficient capture. A separate source of heat would also have to be designed into the gettering surfaces to initially activate them and also to regenerate them.

TABLE II-9. TETF Toroidal Vacuum System Parameters

Gasload (during burn)	60 Torr-l/sec
Pumping speed (during burn)	6×10^6 l/sec

H. Tritium Processing and Containment for the TETF

Essentially all of the tritium entering the torus during operation of the TETF is delivered by the T^0 neutral injectors. Maintaining a feed rate to the torus of ~ 15 gram/hour requires that ~ 120 gram/hour be supplied to the T neutral injection system. Coupling these feed rates with other considerations related to injector pump turn-around times, processor hold-ups and reserve storage requirements has led to an estimated tritium inventory for the TETF of ~ 650 grams. The operating scenario conceived for TETF fuel recycle and processing is based on four T^0 neutral injectors, each equipped with two interchangeable gas pumping systems that are turned over for regeneration on an hourly basis. The tritium inventory within each injector is cleaned up twice a day by one of three parallel exhaust processing systems. Pumping and processing for the four D^0 neutral injectors is carried out in essentially the same manner as described above for the T^0 injection systems. The plasma exhaust is processed by a separate system and is turned around once a day. Each of the three processing systems (for the T^0 injectors, D^0 injectors and the plasma exhaust) contains: 1) a purification train to remove non-hydrogenous impurities and 2) a cryogenic distillation cascade to carry out the required hydrogen isotope separations. Characteristics of the tritium systems are summarized in Table II-10.

Fuel would be stored in a concrete-barricaded vault housing several independently controlled getter-type storage units. The valves, pumps, compressors, monitors and other ancillary components needed for tritium transfer and control operations require only modest development beyond current state-of-the-art hardware.

Selected considerations of safety implications and environmental impacts related to tritium handling in the TETF indicate that: 1) tritium accumulation in the first wall cooling water circuit, 2) a large accidental release amounting to several percent of the total tritium inventory, and 3) a release rate of 100 Curies/day during normal operation, can all be handled within the limitations of current practices and guidelines.

With the exception of several difficult problems associated with the T^0 injector systems, most of the tritium handling and containment technology required for operation of the TETF appears to have direct application or extrapolation to EPRs and other follow-up devices.

TABLE II-10. TETF Tritium System Characteristics

Injection system requirements

T ⁰ injection rate to plasma	15 gm/hr
T ⁰ input rate to injectors	120 gm/hr
Tritium inventory per injector	15 gm

Inventory disposition

Injector systems (4)	120 gm
Processors for injector recycle (3)	45 gm
Plasma exhaust and processing system (2)	360 gm
Contingency reserve storage (1)	<u>120 gm</u>
Total	~ 650 gm

Tritium consumption rate 880 gm/yr

Tritium accumulation in first wall H₂O coolant
after 1 year of operation 10³ - 10⁴ Ci

I. TETF First Wall and Magnet Shield

In order to provide a focal point for the scoping studies of the TETF first wall and shield systems, a preliminary design description of each system was developed based on the following set of requirements: 1) provision of a vacuum environment that would be suitable for sustained operation of a TETF-type plasma; 2) protection of the superconducting TF coils from excessive energy deposition and radiation damage by neutrons and secondary gamma rays; and 3) accommodation of major access penetration for neutral-beam injection, evacuation, coolant inlet and removal, diagnostics, maintenance and experiments. Because breeding of tritium and production of sensible heat are not major objectives of the TETF, no minimum operating temperature is imposed and no efforts were made to include a lithium-containing blanket medium. Instead, the designs for the first wall and shield were based on concepts that would readily lend themselves to fabrication, assembly, operation and maintenance within the limits of existing technology. These designs were then analyzed in sufficient detail to evaluate the validity of the TETF concept, identify major problem areas and uncertainties, and provide a point of departure for a more comprehensive conceptual design study in the future.

The TETF first wall consists of a type 316 stainless steel vacuum vessel made up of twenty-four oval cross section toroidal segments, 2.2 m high and 1.7 m wide. The twenty-four segments are flanged together to form an irregular torus with a major diameter of 6 m. The walls of the chamber are formed from two parallel layers of integrated cooling panel (overall thickness ≤ 2 cm/panel) that are welded to the mating end flanges of each segment and cross braced with lateral spars for additional support. The surface of the inner cooling panel facing the plasma is coated with a low-Z material to minimize degradation of plasma performance by sputtered high-Z materials. Both of the first-wall panel layers are cooled with water and the maximum temperature reached at any point in the vacuum vessel is $< 400^{\circ}\text{C}$. Even in the regions of highest flux the strength properties of the 316 stainless steel structure will be satisfactory for a period in excess of five years. Major materials uncertainties are related to: 1) thermal fatigue resulting from the rather large cyclic thermal stresses imposed by the duty cycle and 2) radiation stability of the low-Z liner material. Stress analyses have shown normal mechanical loadings resulting from the weight of the

cooling panels and the external atmospheric pressure can be accommodated by the proposed structure design with a reasonable safety factor. One-dimensional transient heat transfer analyses of the inner cooling panel shows that adequate heat removal can be achieved in the regions of highest neutron, particle and radiation flux. For a 400°C maximum operating temperature, the temperature drop through a 100 μ m thick graphite coating was $\sim$ 55°C. Temperature gradients radially through and across the surface of the inner stainless steel panel were nominally 150°C and $\sim$ 5°C, respectively. Steady-state cyclic temperature in the first wall was reached after only a few burn cycles; the maximum temperature reached anywhere in the stainless steel was $\sim$ 290°C and the maximum temperature variation or fluctuation during a typical burn cycle was $\sim$ 140°C. Strain variations in the stainless steel resulting from thermal response to the burn cycle were $\sim$ 0.35% for fully restrained cooling panels and $\sim$ 0.15% for panels restrained only against bending.

The TETF shield consists of a series of slab-type blocks surrounding the vacuum vessel. The inner shield blocks (between the vacuum vessel and the interior straight section of the D-shaped TF coil) are composed of stratified layers of stainless steel and B_4C in a configuration designed to maximize the radiation attenuation in a 0.9 m thick zone. The upper, lower and outer shield blocks are composed of a 1.05 m thick zone of stainless steel, lead mortar and a limited amount of B_4C . The latter shield design results in a much lower cost per unit volume than for the inner shield and, hence, the overall cost of the shield system is significantly reduced. With the possible exception of some potential hydrogen embrittlement problems in the lower temperature regions of the shield, materials performance in the radiation environment imposed on the shield should present no serious problems.

Characteristics of the first wall and magnet shield are summarized in Table II-11. The nuclear performance is summarized in Table II-12.

J. Facility Description

The reactor complex consists of eight major facilities covering an area of 40,000 square meters. The focal point is the reactor containment building, a structure 46 m in diameter by 38 m high made of reinforced concrete, 1.5 m thick, to meet both structural and biological requirements. A thin steel membrane, 1 cm thick, lines the inside walls of the building

TABLE II-11. TETF First Wall and Magnet Shield Parameters

First Wall

Material	low-Z coated 316 SS
Thickness	4 cm
Construction	2 layers of integrated coolant panel
Coolant	H ₂ O
temperature (inlet/exit)	38°C/< 100°C
pressure	< 2000 psi
Maximum temperature in SS	290°C
Maximum ΔT through coolant panel	150°C
Maximum ΔT during burn cycle	140°C
Maximum thermal strain variation	
fully restrained	0.35%
restrained against bending	0.15%

Magnet Shield

Material	SS/B ₄ C/Pb-mortar
Thickness	
inside	0.9 m
outside	1.05 m
top and bottom	1.05 m
Coolant	H ₂ O
Maximum structural temperature	< 200°C

TABLE II-12. Summary of the Nuclear Performance of the TETF First Wall and Shield

Average neutron wall loading	0.2 MW/m ²
Assumed duty factor	50%

First Wall

Average fusion neutron current to wall	8.86×10^{12} n/cm ² /sec
Maximum neutron fluence (0-15 MeV)	2.25×10^{21} n/cm ² /yr
Maximum gamma-ray fluence (0-15 MeV)	4.94×10^{20} photon/cm ² /yr
Surface heating (plasma and beam ions, neutrons and radiation)	25 W/cm ²
Maximum nuclear heating	2.1 W/cm ³
Atomic displacement	1.1 dpa/yr
Helium production	21 appm/yr
Hydrogen production	52 appm/yr
Specific radioactivity (curies/cm ³ after 5 years of operation)	20

Maximum Values in TF Coil

Neutron fluence	2.1×10^{16} n/cm ² /yr
Gamma-ray fluence	2.9×10^{15} photon/cm ² /yr
Radiation-induced resistivity in Cu stabilizer (1 MW-yr/m ²)	6.6×10^{-9} Ω-cm
Dose to mylar (1 MW-yr/m ²)	1.0×10^8 rad
Nuclear energy deposition per coil	11 watt
Total nuclear energy deposition (12 coils)	132 watt

forming a barrier in the event of a tritium release. The seal is carried through all penetrations and access ways. A reactor pedestal is provided slightly offset from the center of the building for convenience in positioning the 85 ton overhead polar crane used in assembly and maintenance of the reactor. The building is equipped for remote maintenance of the reactor.

A second containment building, ~ 13 m diameter and 24 m high with a two-story laboratory wing, is provided for tritium storage and fuel reprocessing. This building can be sealed in the event of accidental tritium release. A mockup and maintenance building, 18 m wide, 46 m long and 18 m high with a full coverage 75 ton bridge crane, is located adjacent to the reactor containment and tritium facility. This building contains a 15 m by 8 m hot cell for reactor component repair and experimental services. Damaged neutral beam injectors are transferred directly through an access door from the reactor building. An area for magnet coil winding and assembly along with a full size four-section mockup of the reactor is provided. A power supply facility 24 m by 61 m is located adjacent to the reactor building and houses the bulk of the magnet and cryogenic power equipment. An experimental apparatus area is available in the lower area of this building and in the mockup maintenance building.

A control center consisting of the control room, computer center, laboratories and offices is housed in a four-story building 19 m by 20 m. Other structures include the reactor ventilation building, the cooling towers and the power supply substation.

K. Cost Estimate

The estimated cost of the Tokamak Engineering Technology Facility is \$218 M (1976 dollars). A cost breakdown is given in Table II-13.

TABLE II-13. TETF Cost Summary

		<u>Estimate (\$K)</u>
Structures and Site Facilities		
Site improvements		1,000
Reactor building		10,700
Maintenance/mockup building		6,850
Power conversion building		2,240
Control center		2,300
Tritium processing facility		1,540
Reactor Ventilation building		550
Subtotal		<u>25,180</u>
Reactor		
Reactor vessel first wall		3,770
Reactor shield system		13,750
Reactor vacuum system		7,710
Magnet systems		25,200
Toroidal-field coils	18,160	
Equilibrium-field coils	5,150	
Ohmic-heating coils	1,890	
Neutral beam injectors		12,520
Instrumentation and control		7,000
Reactor support structures		3,120
Subtotal		<u>73,070</u>
Reactor Plant Facilities		
Magnet power supplies		10,600
Ohmic-heating coil supply	3,600	
Equilibrium-field coil supply	4,300	
Neutral beam	2,700	
Reactor coolant system		1,520
Remote handling equipment		2,700
Reactor site power - substation		3,700
Cryogenics		10,000
Tritium handling facilities		9,430
Cooling towers		3,500
Subtotal		<u>41,450</u>
Total		139,700
25% Engineering		<u>34,925</u>
Subtotal		174,625
25% Contingency		<u>43,656</u>
Grand Total		218,281

III. PLASMA PHYSICS AND REACTOR PERFORMANCE

The TETF plasma will be operated in the CIT (counterstreaming-ion torus) operating mode, described in references 1 through 4. (The ideal CIT system is sometimes called the CBT for "colliding beam torus".¹) In CIT/CBT operation, injected neutral beams are used to stack large densities of oppositely traveling D and T ions. As shown in Fig. III-1, in steady state these ions form two distinct, quasi-thermal velocity distributions, oppositely displaced in velocity along the magnetic axis. The relative displacement velocity can be near the optimum value for generating fusion power, which occurs principally by nearly head-on collisions between D and T ions of the counterstreaming distributions. Such distributions are set up by introducing all ions by means of neutral beams injected parallel and anti-parallel to the toroidal magnetic field.⁵ The energetic ions slow down by Coulomb drag on the plasma electrons, and if completely decelerated ions are removed from the plasma in a time that is short compared with the fast-ion slowing-down time, virtually all plasma ions will be found in the toroidally-streaming distributions.

A. Ideal CIT Performance

A low-temperature target plasma, with electron density $n_e \sim 2 \times 10^{13} \text{ cm}^{-3}$, is required to initiate CIT operation. After beam injection for several slowing-down times (typically, less than 0.5 s), the electron temperature T_e reaches a steady-state value and the ion population is dominated by energetic ions -- provided that the inward flow rate of cold particles from the wall region to the reacting region is kept small compared with the rate of particle injection by the beams. In steady state, injected energetic neutrals are trapped by charge-exchange and impact-ionization collisions with the ions in the energetic distributions.

Energetic ions are assumed to be perfectly confined until they decelerate to an energy $W \sim 2T_e$. Ions with energy $W < 2T_e$ are assumed to be lost at a rate $\tau^{-1} \propto \tau_E^{-1}$, where τ_E is the electron energy confinement time. Fokker-Planck analysis¹ has shown that Coulomb collisions among ions of the same beams cause significant energy diffusion during the slowing-down process, while interaction between ions in opposite beams tends to produce angular scattering. As a result of these processes, the ions in each beam tend to

Figure III-1. Distribution functions for the D and T velocity components parallel to the toroidal magnetic field in a CIT/CBT plasma. D^0 injection energy = 60 keV, T^0 injection energy = 40 keV. (In practice, the T^0 would be injected at 90 keV.) Velocity units are relative. (Taken from reference 1.)

form a "thermal" distribution, with a mean velocity in the laboratory frame somewhat less than the injection velocity.

Calculated distribution functions $f_i(v)$ for typical injection energies are shown in Figs. III-1 and III-2. The ion "temperature" is about 14 keV, or approximately $2T_e$. The electron temperature is maintained almost entirely by power flow from the fast ions, and is given by the steady-state power balance relation

$$\frac{3}{2} \frac{n_e T_e}{\tau_E} = \frac{C n_e \ln A_e}{T_e^{3/2}} \sum_{D,T} \frac{n_i (\bar{W}_i - 3/2 T_e)}{A_i} \quad (1)$$

where $C = 1.06 \times 10^{-13}$ (T_e, W in keV), n_i is the density, A_i the atomic mass, and $\bar{W}_i$ the average ion energy in each "beam". In practice, T_e and n_e are determined by τ_E and the energetic-ion source strength. Figure III-3 shows the T_e that is obtained with given values of $n_e \tau_E$ and injection energy, W_0 . The ion lifetime τ_h , is given by

$$\frac{\tau_h}{\tau_E} \approx \frac{(W_0 - 3/2 T_e)}{3/2 T_e} \gg 1 \quad (2)$$

That is, the ion lifetime (during slowing down) must be several times larger than the electron energy confinement time. However, the lifetime of decelerated ions should be as short as possible, in order that the bulk of the plasma ions can be found in the counterstreaming distributions, and the fusion reactivity maximized. The distribution functions of Figs. III-1 and III-2, and the ideal performance discussed in the rest of this section, are calculated assuming that the lifetime of decelerated tritons and deuterons is no greater than τ_E^{1-3} .

The stability of the ideal CIT velocity distributions has been examined for electrostatic and electromagnetic modes both near and far from the ion cyclotron frequency.¹⁻⁶ For practical values of W_0 , T_e , and confinement times, all velocity-space modes are stable. When nonideal effects are taken into account (see Section III-B), the ion velocity distributions become even broader,

Figure III-2. Steady-state contours of D and T velocity distributions in a CIT/CBT plasma. The deuterons are injected at 60 keV, $\theta = 4^\circ$ and the tritons at 40 keV, $\theta = 176^\circ$. Velocity units are relative. (Taken from reference 1.)

Figure III-3. Dependence of T_e on electron energy confinement parameter for steady-state CIT operation. Deuterons and tritons are injected at energy W_0 . (Taken from reference 3.)

and the directed velocities are reduced so that velocity-space stability is enhanced.

As far as finite-medium modes are concerned, the plasma conditions of interest fall into the trapped-electron regime. The displaced-velocity distributions should have no effect on the amplitude of the trapped-electron mode.⁷ Theory predicts that the particle diffusion rate caused by this mode is many times less than the energy diffusion rate, so that if τ_E is determined in large part by this instability, it is reasonable to expect that $\tau_h \gg \tau_E$, as required for CIT operation.

The fusion reactivity of the ideal CIT system can be larger than the maximum reactivity of a thermonuclear plasma (namely, $8.8 \times 10^{-16} \text{ cm}^3/\text{s}$ at $T_i = 60 \text{ keV}$), and it approaches that of a beam-target (TCT) system requiring 3 times the injection voltage.^{2,3} For a given T_e , fusion power multiplication Q of the CIT can be somewhat larger than that of a TCT.¹⁻⁴ (Here we calculate Q by integrating fusion reactivity over the D and T velocity distributions and dividing by the injected power.) Figure III-4 shows that for injection energy of 60 keV, $Q = 1$ (break-even) is attained, ideally, at $n_e \tau_E \approx 1.5 \times 10^{13} \text{ cm}^{-3}\text{s}$ and $T_e = 3.3 \text{ keV}$, with $n_e \tau_h \approx 1.5 \times 10^{13} \text{ cm}^{-3}\text{s}$. The fractional burn-up of tritium per pass increases rapidly with T_e , but rather slowly with W_o .² At $W_o = 60 \text{ keV}$, f_B is 1.1% at $T_e = 4 \text{ keV}$, and 3.9% at $T_e = 12 \text{ keV}$.

Figure III-5 shows the 14 MeV neutron production rate from ideal CIT plasmas of circular cross section as a function of plasma radius, a , and W_o , when $n_e \tau_E$ is such that $T_e = 4.0 \text{ keV}$. For a given a , the neutron output is proportional to $n_e^2 \langle \sigma v \rangle$. At small W_o , n_e is determined by the neutral-beam penetration requirement, which we take as trapping length, $\lambda_t = a$. At large W_o , n_e is limited by the allowed plasma pressure, which we take to satisfy $\beta_p = 2/3$. Therefore, for each radius there is an optimal W_o for maximum neutron output. Evidently, for tokamak plasmas of PLT size ($a = 40\text{--}50 \text{ cm}$), and using beam voltages at which high-power neutral beams are now available ($W_o \sim 40 \text{ keV}$), neutron intensities in the range $10^{18}\text{--}10^{19} \text{ s}^{-1}$ are attainable.

In addition to the use of low-energy neutral beams and the relatively lenient confinement requirements for significant neutron production, an advantage of CIT operation is that all fueling is performed by the injected beams.

Figure III-4. Ideal fusion power gain, Q , for steady-state CIT operation. Deuterons and tritons are injected at the same energy. 17.6 MeV per reaction. (Taken from reference 3.)

Figure III-5. Production rate of 14 MeV neutrons from D-T counterstreaming-tokamak plasmas of circular cross section. Deuterons and tritons are injected with the same energy. $B_t = 4.0 \text{ T}$, $A_0 = 5$, $q = 2.5$ and $\beta_p \leq 3.3$. (Taken from reference 3.)

In fact, if this condition is violated, so that an appreciable fraction of the particle influx is due to neutral recycling, for example, the reactor performance deteriorates, as discussed in the next section.

B. Nonideal Effects

1. Neutral Influx

The degree to which ideal CIT performance can be attained depends most crucially on minimization of the warm-ion population. The proportion of warm ions depends, first, on the residence time, τ_p , of decelerated ions in the reacting region, and second, on the influx of neutrals and cold plasma to this region. As discussed earlier, it seems likely that $\tau_p \lesssim \tau_E$, since the lifetime of injected ions during their slowing-down period is already at least several times τ_E , and these fast ions diffuse outward before complete deceleration.^{2,3} The most severe limitation on CIT performance, therefore, is expected to be set by the influx of neutrals and cold plasma.

For $W_0 = 40$ to 80 keV, about half the injected energetic atoms are trapped by charge exchange.² Each of the resulting neutrals undergoes several charge-exchange collisions, or is ionized, before it can escape. This process results immediately in a loss of 5 to 10% of the injected power. The steady-state density of neutrals in the reacting region due to this process is $n_{nb} \sim 10^{-6} n_e$, which does not significantly affect the reactor performance.

A much more serious problem is the influx of neutrals from the scrape-off region surrounding the discharge. For $\langle n_e \rangle a \gtrsim 2 \times 10^{15} \text{ cm}^{-2}$, where a is the plasma radius, one expects the neutral density to be attenuated by at least a factor of 1000 between $r = a$ and $r = 0$.⁸ If for optimal performance we require that the contribution of this process to the total neutral density in the reacting region be no larger than n_{nb} , then the neutral density at the plasma edge must be $n_{na} \lesssim 10^3 n_{nb} \lesssim 10^{-3} n_e$. This relation determines the vacuum pumping requirement:

$$1/4 n_{na} \bar{v}_n A_p = F_b + F_g.$$

Here A_p is the effective "black-hole" pumping area of the torus, F_b is the injected beam-atom flux, and F_g is the slow-gas influx into the torus from the

beam ducts. One can maximize either $\bar{v}_n$ or A_p in order to minimize n_{na} . One method of achieving high-velocity outflow is by use of a poloidal magnetic divertor,⁹ that exhausts from the torus those ions diffusing out of the reacting plasma ($\bar{v}_n > 10^7$ cm/s). But provision for a magnetic divertor is a complex and costly affair, and divertor effectiveness remains to be demonstrated. At this time, it seems more prudent to attempt to maximize A_p .

One method of maximizing A is the installation of gettering surfaces on the torus wall, but this procedure seems impractical for a reactor. One possible solution is to cover about 10% of the torus wall with baffled apertures that separate the torus from large pumping chambers with (baffle-protected) cryosurfaces, or gettering surfaces. It should be emphasized that the "temperature" of the neutral gas in the scrape-off region is expected to be about 10 eV,⁸ so that $\bar{v}_n \sim 3 \times 10^6$ cm/s.

The effects of the slow-neutral influx are to enhance charge-exchange loss of energetic ions and to increase the proportion of warm ions in the reacting region. The population of warm ions in this region is increased also by diffusion from the surrounding region of colder plasma. (In calculating Q , this inward diffusion can be modeled most conveniently by using larger values of τ_p/τ_E .) Figure III-6 shows how Q decreases with increasing warm-ion population, when $W_0 = 60$ keV (D^0), and the injected beam power is constant. With increasing warm-ion flux, beam-target reactions become more important and eventually become the dominant source of neutrons. In the limiting case of interest, the plasma is operated in the TCT regime, driven and fueled by injected D and T beams.¹⁰ In this TCT regime, the Q -value, and the neutron production for a given beam power (for $W_0 \sim 60$ keV), are reduced by a factor of 5 to 7 from their values in ideal CIT operation. Furthermore, $n_e \tau_E$ must be increased to maintain the same T_e .

In our reference design (Section III-D), we assume that the inward flux of warm plasma can be kept sufficiently small so that effectively $\tau_p/\tau_E = 1.5$. When operating at $T_e = 4.5$ keV, $W_0 = 60$ keV (D^0), and taking into account all charge-exchange losses, Q is reduced to about 45% of its ideal value.

2. Impurity Effects

The principal effect of impurity ions in the CIT is to enhance pitch-angle scattering, which undermines the advantage of head-on collisions.

Figure III-6. Dependence of fusion power gain on relative fluxes of warm ions and hot ions (supplied by neutral beams) to the reacting region. $Q_{\max}$ is the ideal gain for CIT operation. The limiting power gain for a TCT plasma at the same temperature (with D and T injection) is indicated.

Evidently, this effect is less important when thermal ions make up a substantial fraction of the total ion population. A second deleterious effect is the additional Coulomb drag on energetic ions by the impurity ions and their neutralizing electrons. Impurity radiation losses tend to make up only a small fraction of the total power balance in this highly beam-driven plasma.

On the other hand, a small impurity population (such as $Z_{\text{eff}} \sim 1.5$) has the beneficial effect that it enables the streaming energetic ions to carry a net current that can be a substantial fraction of the total plasma current.²

C. Machine and Plasma Parameters

A vital consideration is the choice of injection energy. In ideal CIT operation, maximum Q is obtained in the range $W_0 = 40\text{--}50$ keV, when $T_e = 3\text{--}10$ keV.^{2,3} However, there are several practical reasons for preferring somewhat larger beam energies: 1) the attainable neutron intensity increases with W_0 (Fig. III-5), although the corresponding plasma size must also increase; 2) if a substantial warm-ion population is present, at large W_0 one can take better advantage of target-plasma reactions, so that Q is not as severely degraded; 3) charge-exchange energy loss decreases with increasing W_0 ; and 4) if $Z_{\text{eff}} > 1$, angular scattering is reduced at larger W_0 .

With the use of direct energy recovery, satisfactory injector efficiency ($\sim 40\%$) can be attained with beam energy up to about 100 keV for D^0 and 150 keV for T^0 , using present-day ion sources with separation of molecular-ion components.¹¹ Given a desired neutron intensity and the available magnetic field, the size of the plasma and the optimal W_0 are fairly closely determined. In the present design, where a neutron flux close to $10^{13} \text{ cm}^{-2}\text{s}^{-1}$ is desired, the axial field is 4.0 T, and the plasma can be somewhat elongated, the beam energies are chosen as 60 keV for D^0 and 90 keV for T^0 . (90 keV T^0 beams can be produced with the same efficiency as 60 keV D^0 beams and the beams have equal penetration lengths in the plasma.)

The TETF geometry is summarized in Table II-1. The dimensions have been chosen to give the smallest possible device that provides a peak neutron flux close to $10^{13} \text{ n/cm}^2/\text{s}$, when Q is 45% of its ideal value (Section III-B). The major radius of the plasma is determined by the minimum allowed thicknesses of the transformer core (Chapter IV), the toroidal-field coils (Chapter IV), and the shielding required to protect these coils (Chapter VIII).

Most of the thermal output of the reacting plasma is removed by the water coolant that flows in the shielding region (Chapter VIII). Inside the first wall, a plasma scrape-off channel of 20 cm thickness is specified. As a result of these considerations, the inner edge of the discharge column is located at $R = 2.45$ m.

Under the constraints of maximum plasma pressure and adequate beam penetration [e.g., $\beta_p \propto 2/3(R_0/a)$ and $\lambda_t \sim a$], higher neutron wall loading is possible by employing a vertically elongated (noncircular) plasma cross section. The equilibrium and stability of elongated tokamak plasmas operating at low q and high β_p is still a matter of uncertainty.¹² However, it does appear that a vertical elongation of $b/a = 1.5$ at $q = 3$ is feasible, and recent experiments along these lines have proven encouraging.¹³ Consequently, our design specifies a vertical elongation as large as 1.5 with a D-shaped cross section (shape factor = 1.44).

The plasma radius, $a = 0.55$ m, has been chosen so that $\lambda_t \sim a$ when the total plasma pressure corresponds to $\beta_p \sim 2/3(R_0/a)$ at $q = 3.0$ and $b/a = 1.5$. Other plasma parameters are shown in Table III-1. The maximum field at the NbTi windings is nominally 9.6 T, although leakage fields and stray vertical fields can increase the total field strength to about 10 T. A method of operating up to 10 T without pumping directly on the helium bath is described in Chapter IV.

In reference design A of Table III-1, $\lambda_t/a = 0.9$, so that considerable on-axis peaking of the neutral-beam deposition profile is obtained.¹⁴ This peaking ensures that for neutral densities outside the discharge $\lesssim 2 \times 10^{11}$ cm⁻³, the ion population, over a large central region of the plasma, is dominated by energetic ions.

The total injected neutral-beam power required to maintain the conditions shown is 30 MW. The production rate of fusion alpha particles is 4.2 MW, and about 50% of these alphas thermalize in the plasma.

If the maximum field at the coils is limited to 8.0 T (case B of Table III-1), then the plasma pressure must be reduced by 30%. Only 21 MW of injected power are required to maintain the plasma, but the 14 MeV neutron production rate is only 62% as large as in case A.

Finally, case C of Table III-1 gives the relevant plasma parameters for $B_{\max} = 10$ T when the warm-ion influx is so large that the plasma must be

TABLE III-1. TETF Plasma Parameters

	Case A	Case B	Case C (TCT)
R_o (m)	3.0	3.0	3.0
a (m)	0.55	0.55	0.55
A	5.45	5.45	5.45
Vertical elongation (maximum)	1.5	1.5	1.5
Horizontal wall radius (m)	0.75	0.75	0.75
Plasma volume (m ³)	30.7	30.7	30.7
Wall area (m ²)	131	131	131
B_t at coil (T), ($R = 1.25$ m)	9.6	8.0	9.6
B_t on axis (T), ($R = 3.0$ m)	4.0	3.3	4.0
I_p (MA)	1.40	1.17	1.4
q	3.0	3.0	3.0
$\langle n_e \rangle$ (cm ⁻³)	4.0×10^{13}	3.1×10^{13}	5.3×10^{13}
$\langle T_e \rangle$ (keV)	4.5	4.0	4.5
$\langle n_e \rangle \tau_E$ (cm ⁻³ s)	2.6×10^{12}	2.0×10^{12}	5.8×10^{12}
W_o (keV) D ^o	60	60	60
W_o (keV) T ^o	90	90	90
Beam power (MW) D ^o	15	10.5	15
Beam power (MW) T ^o	15	10.5	15
Beam mean-free path (m)	0.49	0.63	0.37
β_p (excluding alphas)	3.6	3.6	3.6
Q	0.70	0.62	0.23
P_f (W/cm ³)	0.68	0.42	0.22
Fusion neutron power (MW)	16.8	10.4	5.5
Total thermal power (MW)	51	34	37

operated in the TCT regime. While the maximum plasma density allowed for adequate penetration of the beams in TCT operation (say, $\lambda_e/a \geq 1/2$) is $\langle n_e \rangle \sim 7 \times 10^{13} \text{ cm}^{-3}$, at this value only a low plasma temperature is possible because of the plasma pressure limitation. A larger T_e is preferred in order to enhance Q . For a given T_e , the required $n_e \tau_E$ increases with decreasing Γ = hot-ion pressure/bulk-ion pressure. Taking $\Gamma = 0.65$ and $T_e = 4.5 \text{ keV}$ (as in Case A), we find that $\langle n_e \rangle = 5.3 \times 10^{13} \text{ cm}^{-3}$. The required $n_e \tau_E$ is significantly larger than in the CIT case but still appears reasonable. The total neutron production rate is only one-third that of Case A.

1. Vacuum Pumping Requirement

If the pressure in the beam ducts can be maintained at 10^{-5} torr, then using the notation of Eq. 3 we have $F_b = 2.6 \times 10^{21} \text{ s}^{-1}$ and $F_g = 1.3 \times 10^{21} \text{ s}^{-1}$. Then $n_{na} = 5.2 \times 10^{15}/A_p$ where A_p is in cm^2 . For the conditions of Case A of Table III-1, we require that $n_{na} \sim 10^{11} \text{ cm}^{-3}$ so that $A_p \sim 5.2 \times 10^4 \text{ cm}^2$. This requirement is met if 10% of the torus wall is covered by pumping apertures that have capture efficiency $\gamma_p = 40\%$. If the efficiency is smaller than this value, the reactor performance will degrade toward the parameters of Case C. Methods of achieving γ_p of several tens of percent are discussed in Chapter VI.

2. Fusion Alpha Particles

The confinement of fusion alphas is not necessary for CIT operation, although their energy deposition in the plasma does help performance. Confined alphas give up their energy entirely to the electrons, thereby allowing a larger T_e (and Q) to be obtained with a given $n_e \tau_E$. For our reference case, about 50% of the alphas thermalize in the plasma so that the required $n_e \tau_E$ for $T_e = 4.5 \text{ keV}$ is reduced by 7%. Evidently, the contribution of alphas to the total plasma pressure can be neglected.

3. Beam-induced Toroidal Current

The counterstreaming-ion distributions have currents associated with them. It is easily shown that when $Z_{\text{eff}} = 1$, the total plasma current is zero unless an external electric field E is applied.^{2,15} But if

the plasma contains an impurity population or if a significant fraction of the electrons are trapped in banana orbits or in toroidal-field ripple, then $Z_{\text{eff}} > 1$ and a toroidal current can be maintained even when $E = 0$. The current density that can be sustained by the beams is approximately

$$J = e |n_T \langle v_{T11} \rangle - n_D \langle v_{D11} \rangle| (Z_{\text{eff}} - 1) / Z_{\text{eff}}. \quad (4)$$

For the reference conditions described above, $J \sim 40 \text{ A/cm}^2$ when $Z_{\text{eff}} = 1.5$. This value is about one-half the total current density in our reference design A (see Section III-C).

In the outer region of the plasma (say, $r/a \gtrsim 2/3$), a significant Z_{eff} is established by the trapping of electrons in banana orbits and in field ripple. However, it appears best to minimize the energetic-ion population in this region because of enhanced charge-exchange losses. Therefore, the beam-induced current is more likely to be found in the central region of the plasma and depends on a small impurity content. In practice, one can expect that Z_{eff} will be at least slightly larger than unity and that the momenta of the D and T will never quite cancel so that some net current will always be driven by the counterstreaming ions. Obviously, one must adjust the relative neutral-beam intensities so that this current flows in the same direction as that driven by the OH transformer; the demands on the latter are reduced correspondingly. However, in calculating the volt-sec requirement of the OH transformer (see Chapter IV), we do not take credit for any plasma current that is contributed by the streaming ions.

D. Burn Cycle

Since all plasma fueling is carried out by the injected beams, the length of the burn period is determined either by impurity buildup or by the volt-sec available in the transformer core to drive the toroidal current. While the streaming ions are, in fact, capable of carrying a large portion of the toroidal current, the reference design specifies sufficient volt-sec so that the transformer can by itself support a burn time of at least 1 min., with Z_{eff} up to 2 (see Chapter IV). The sequence of events during a burn cycle is summarized in Table III-2.

TABLE III-2. Burn Cycle Sequence

Inject gas and form target plasma at $I_p \sim 500$ kA	0 - 0.5 sec
Begin injection, raise I_p to 1400 kA, attain equilibrium plasma	0.5 - 2 sec
Burn (continuous injection)	2 - 44 sec
Shut-off beams and ramp-down plasma current	44 - 46 sec
Exhaust torus to 10^{-6} torr and charge OH transformer primary	46 - 60 sec

The burn cycle of Table III-2 gives a duty cycle of 70%.

E. Radiation Source Characteristics

Table III-3 lists the neutron source characteristics for the three cases of Table III-1. The increase in neutron current at the horizontal midplane on the outside is due to geometrical effects and the outward shift in the plasma pressure profiles.¹⁶ In all cases, the low neutron current precludes the study of bulk radiation damage at high damage levels. However, the surface radiation flux (D^+ , T^+ , line radiation, etc.) is sufficient for obtaining sputtering and blistering information in a reasonable period of time. The power flow to the vacuum wall in the form of particles and radiation is listed in Table III-4. This flow of 0.26 MW/m^2 (averaged over the first wall) is the same as that deposited on the first wall of an ignition reactor with a neutron wall loading of 1.0 MW/m^2 (although the distribution of power among the various radiation components would be somewhat different). The flux of particles of a given type depends on the energy distribution of these particles as they escape from the plasma, and these distributions cannot be accurately determined until elaborate transport calculations are performed.

TABLE III-3. 14-MeV Neutron Source Characteristics

	Case A (CIT)	Case B (CIT)	Case C (TCT)
Neutron intensity (n/s)	7.3×10^{18}	4.5×10^{18}	2.4×10^{18}
Neutron power (MW)	16.8	10.4	5.5
Average wall loading (MW/m ²)	0.13	0.08	0.04
Wall loading at midplane, R = 3.75 m (MW/m ²)	0.16	0.10	0.05
Average current (n/cm ² /s)	5.7×10^{12}	3.5×10^{12}	1.9×10^{12}
Current at midplane, R = 3.75 m (n/cm ² /s)	7.1×10^{12}	4.4×10^{12}	2.3×10^{12}

TABLE III-4. Power Flow to Wall

Injected Power = 30 MW 14-MeV Neutron Production = 16.8 MW Total Non-neutron Power = 34.2 MW			
Radiation Type	Power Flow (MW)	Energy Range	Flux (particles/ cm ² /s)
Unconfined fusion alphas	2.2	3-3.5 MeV	3.0×10^{12}
Charge-exchange neutrals	6	10-2000 eV	$\sim 1 \times 10^{17}$
Electron heat cond. and par- ticle flow	14	10-100 eV	$\sim 2 \times 10^{18}$
Line rad., brems., synch. rad.	2	0.01 eV to 100 keV	—
Ion heat cond. and par- ticle flow	10	10-100 eV	$\sim 1 \times 10^{18}$

F. Performance Testing on PLT and PDX

The PLT and PDX tokamaks at PPPL are similar in size to the TETF reference design (but with much smaller aspect ratios) and each device will be equipped with about 4 MW of co- and counter-injected 40 keV neutral beams. The injection pulses will be sufficiently long ($\gtrsim 300$ ms) so that plasma conditions will reach a steady state. During the next few years, these devices will provide an opportunity to test the feasibility of the most crucial features of CIT performance: 1) that recycling of neutrals from the wall can be reduced sufficiently so that, in the central reacting region, most of the plasma ions are introduced by the energetic beams; 2) that the lifetime of energetic ions can be several times τ_E ; and 3) that Z_{eff} can be kept below about 2. The ability of the energetic ions to maintain a substantial fraction of the plasma current can also be investigated.

In these experimental devices, it will be possible to determine accurately the ion velocity distributions by means of charge-exchange and (D-D) neutron-energy-spectrum techniques. The Q-values measured with D-D can be extrapolated readily to a D-T system.

Evidently, the crucial test will be the degree of attainment of condition (1), which determines whether the device must operate in a TCT mode (with relatively low Q) or whether it can sensibly approach the CIT regime. In order that condition (1) be remotely satisfied in PLT, effective gettering of the torus wall will be essential. (Operation of this kind was performed successfully on the ATC device,¹⁷ where a large fraction of the exiting plasma was captured on the wall and Z_{eff} was maintained below 2.) However, prospects for success are much greater on the PDX device, which is especially designed with elaborate getter-pumping surfaces housed in large compartments above and below the discharge region (somewhat as envisaged in the preferred option for the present design, see Chapter VI). Furthermore, the effectiveness of a poloidal magnetic divertor in pumping ions that diffuse out of the discharge can be tested on PDX. The success of this latter technique would provide another option for maximizing reactor performance.

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IV. MAGNET SYSTEM

This chapter covers the design scoping studies of the superconducting toroidal field (TF) coils, the Ohmic heating (OH) coils and the equilibrium coils for the Tokamak Engineering Technology Facility (TETF).

The TF coils must be superconducting in order to avoid huge power consumption and a high-cost power supply. This superconducting TF coil system will also serve for developing TF-coil engineering technology. NbTi is the most suitable superconductor to use because of its great ductility. It is desirable although not essential that the toroidal-field level be as high as 10 T in order to get optimal performance for the TETF. To attain a toroidal field level higher than about 8.5 T, one must use either Nb₃Sn at 4.2°K, or NbTi operated at a temperature below 4.2°K. Glass-like brittleness of Nb₃Sn, however, sets a severe limitation on its use in large-sized magnets such as the TF coil for the TETF. We have chosen to achieve the 10 T peak field with NbTi operated at about 3°K. A novel 3°K refrigeration system is proposed which will be more economical than the usual vacuum pumping system. The minimum goal of the TF-coil system will be 8 T peak field at 4.2°K operation. This produces a toroidal field of 3.25 T at the plasma center. The maximum goal of the TF coil system is 10 T peak field at 3°K operation, which will generate a 4.06 T toroidal field at the plasma center.

The required equilibrium field (EF) for the TETF is .28 T based on the low- β_0 circular plasma model. The equilibrium field coils were carefully arranged so that the vertical field satisfies the plasma stability requirement for a low- β_0 machine. The exact equilibrium field for a high- β_0 and noncircular plasma requires full understanding of the magnetic surfaces within the plasma, which has not been done in this preliminary study. The EF coil is placed outside the TF coil so that coil assembly and disassembly, blanket and shield assembly and disassembly, the first wall assembly and disassembly will be much easier. The EF coils are decoupled magnetically from the ohmic heating coils and yet produce a bonus of 7.8 volt-sec for the plasma, thus reducing the volt-sec requirement for the ohmic heating coils. It is not clear that ac superconducting coil technology will be developed in time for the TETF fabrications; for this reason, a water-cooled copper EF coil is designed, with a superconducting EF coil system design as an option.

The OH coils are located outside the TF coils and arranged so that they impose a minimum field and torque on the TF coils. It is also necessary that the OH coil field in the plasma-column region be nearly zero so that the plasma can be initially ionized. It is estimated that the total volt-sec required during the startup period is 12, of which 4.2 volt-sec must be supplied by the OH coil. During the burn cycle, the OH coil is required to deliver 5 volt-sec to maintain the plasma current over steady burn period as long as 1 minute.

A. Rationale of the TF Coil Design

As shown in Fig. IV-1a, the TF coils must be large enough to contain the 1.1 m wide plasma, 0.2 m plasma scrape-off region, the first wall, a minimum thickness of blanket and shield for superconducting TF coil protection, and a test sample region of at least 20 cm between the first wall and the shield on the outside. The shield is made thinner on the inside, to maximize the toroidal field level at the plasma. To keep the nuclear heating of the superconducting coils at a tolerable level, and the induced resistivity of the copper stabilizer in the reasonable range, a minimum of 90 cm is necessary without using very expensive shield material. Access requirements for the neutral beam injectors and for maintenance equipment between the TF coil and the blanket on the outside, together with space requirements for the toroidal vacuum vessel and the shield, led to a choice of horizontal bore = 3.65 m for the TF coil. The radial location, R_{10} , in Fig. IV-1a, was determined by the requirements that the flux core of the ohmic heating (OH) coils be large enough to provide the necessary flux swing to start the plasma and to drive the plasma without excessive fields in the OH coil winding.

Field ripple due to the gaps between TF coils can cause increased plasma energy loss, although this process is rather unimportant in the present reactor. We require, arbitrarily, that field ripple be kept less than 1%. To achieve a 1% field ripple, one can either increase the number of TF coils or extend the horizontal field bore of the TF coil. It is undesirable to have to disassemble such large superconducting coils in order to gain access. With these requirements, and using a set of 12 coils, a pure tension coil shape was determined. Fig. IV-1a and Fig. IV-1b illustrate the TFC design and Table IV-1 lists the specifications and magnet characteristics of the ANL-TETF coils system.

Figure IV-1a. Schematic of TETF magnets design

NUMBER OF COILS	= 12
PEAK FIELD	= 100 KG.
$B_{\theta 0}$	= 40 KG.
J AVG.	= 2280 AMP/CM ²
CROSS-SECTION/COIL	= 2174 CM ²
AMP.-TURNS/COIL	= 4.96×10^6 A-T
TOTAL AMP.-TURNS	= 59.52×10^6 A-T
FIELD RIPPLE E_0	= 0.99 R
I	= 10,000 AMP

Figure IV-1b. Schematic plan cross-section of TF coils.

TABLE IV-1. Superconducting TF Coil Characteristics

Superconductor/Stabilizer/Insulator/Support	NbTi/Cu/Fiberglass Epoxy/SS
Number of Coils	12
Coil Shape	Pure-tension "D"
Field Ripple	0.99%
Maximum Access	2 m
Operating Temperature	3°K/4.2°K
Peak Field	10 T/8 T
Superimposed ac Field	0.2 to 0.4 T
Vertical Bore	6 m
Horizontal Core	3.65 m
Field in Plasma	4 T/3.3 T
Operational Current	10,000 A/8,000 A
Stored Energy (10 T)	3100 MJ/total, 258 MJ/coil
Inductance	61 H/total, 5.08 H/coil
Ampere-turns (10 T)	59.52 MA/total, 4.96 MA/coil
Coil Winding Cross Section	1872 cm² (trapezoidal)
Average Current Density (10 T)	2650 A/cm²
Mean Turn Length	18 m
Mean Cross Section per Turn	3.774 cm²
Approximate Weight	35 tons/coil, 420 tons/total
Coil Assembly Thickness	
Outer Cryogenic Shield	0.05 m
Support Cylinder	0.17 m
TF windings	0.40 m
Inner Cryogenic Vacuum and Shield	0.10 m

B. TF Coil Design

1. Pure Tension Coil Shape and Coil Stress

Large nonuniform Lorentz forces are generated in the coils subjecting the TF coil to large bending moments in addition to a large effective internal pressure. To minimize the coil structural requirements, the TF coil must be in pure tension with no bending moments.

ANL has developed the pure tension coil theory and a computer code for designing the practical pure tension TF coil system in which there are access gaps between two adjacent coils. Three dimensional stress analysis has verified the Argonne pure tension coil has uniform tension throughout the TF coil.

A cryostatically stable magnet design is the only viable design.

2. Nuclear Heating and Radiation Damage

With 90 cm inner shield design, the maximum nuclear heating in the magnet conductor is 1.5×10^{-5} watt/cm³, which generates a temperature rise of only .001°K. The total nuclear energy disposition is 11 W per TF coil and 408 W for 12 TF coils. The maximum radiation induced resistivity in the copper stabilizer is 6.60×10^{-9} Ω-cm. This will require an additional 20% of copper for full stabilization. The induced resistivity decreases to 4×10^{-10} Ω-cm at a radial depth of 40 cm in the magnet winding. The radiation induced resistivity in the copper can be recovered by annealing at approximately 300°K.

Fiberglass epoxy laminate will be used as the conductor insulation. Fiberglass can sustain radiation damage up to an integral wall loading of 10 MW-yr/m², which is well in excess of the value to be obtained in the TETF.

3. Conductor Design and Coil Winding

Since the toroidal field decreases rapidly across the coil winding, the conductor is made in 9 grades with peak field specifications at 10.5 T, 10 T, 9 T, 8 T, 7 T, 6 T, 5 T, 4 T, and 3 T, respectively. The innermost layer conductor has the largest amount of NbTi and copper stabilizer because of the highest induced resistivity for the innermost layer conductor. Grading

the conductor in this manner results in a substantial savings in conductor cost and a substantial reduction in the hysteresis loss of NbTi. Typical conductor cross sections are specified and listed in Table IV-2.

The proposed coil winding scheme is helical layered as shown in Fig. IV-2. This coil winding structure will provide good overall mechanical integrity for the TF coil, which is very important because of the overturning moments (the out-of-plane load). The helical layered winding is also especially suitable for the trapezoidal coil cross section.

The conductors are cabled since a monolithic conductor would have excessive ac losses and helium consumption from the pulsed poloidal fields. Type 316 stainless steel is used as the hoop stress supporting members within the conductor. The ac superimposing field consists of longitudinal $\dot{B}_{11}$ of order 0.2 T/sec and a transverse $\dot{B}_1$ of order 0.2 T/sec. The orientation of the stainless steel and the braided cable is such that eddy current loss is minimized. A suitable cable design for this configuration is a braided cable such as that used in the Isabelle dipole magnet in the Brookhaven National Laboratory. A typical 10,000 ampere - 10 T conductor cable is shown in Fig. IV-3. The filaments of NbTi are arranged in a circular ring at equal distance from the conductor strand axis so that eddy currents induced by $\dot{B}_{11}$ will be eliminated. A typical strand is also shown in Fig. IV-3. Within each strand, a sufficient amount of stabilizing copper is incorporated to fulfill cryostatic stabilization criterion. High resistivity soft solder must be used to solder the strand together and thus improve the stability of a braided cable. A stainless steel band is used to bond braided cables and stainless steel supports together. Wetted G-10 fiberglass epoxy is used to further band the cable together. The spiral wrapping G-10 provides turn-to-turn insulation as well as non-vapor-locking liquid helium channel for a TF coil in a standing configuration. Additional cooling passages are also provided in the stainless steel support to permit liquid helium penetration into the braided cable.

TABLE IV-2. Typical Conductor Cross Section of 10 T TF Coil

Magnetic Field (Tesla)	10	9	8	7	6	5	4	3
NbTi Critical Current Density (kA/cm ²)	70	100	130	155	180	195	210	250
NbTi Design Current Density (kA/cm ²)	50	70	93	110	140	155	170	210
NbTi Cross Section (cm ²)	0.2	0.14	0.11	0.09	0.07	0.065	0.059	0.048
Stabilizer Cross Section (cm ²)	2.0	1.4	1.1	0.9	0.7	0.65	0.6	0.5
Liquid Channel and Insulation Cross Section (cm ²)	0.945	0.945	0.945	0.945	0.945	0.945	0.945	0.945
Stainless Steel Cross Section (cm ²)	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
Total Cross Section								
Per Turn (cm ²)	4.45	3.85	3.55	3.35	3.15	3.1	3.05	2.95
Turn Width (cm ²)	2.96	2.56	2.36	2.23	2.1	2.06	2.03	1.96
Turn Thickness (cm ²)	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

Figure IV-2. Toroidal-field coil outer leg vessel

Figure IV-3. A typical toroidal-field superconducting cable (10,000 amperes at 10 T).

Braided cable will not sustain mechanical hoop stress to any appreciable extent and, as a result, the hoop stress must be carried by stainless steel alone. Therefore, stainless steel can be utilized at higher stress levels without affecting the strain level of the copper stabilizer. Using a safety factor of 2, the design stress of stainless steel is 45,000 p at 4.2°K. Since the radius of curvature of the 10 T TETF TF coil is 0.96 m at the peak field location, for a 10,000 ampere cable conductor, the require average cross section for stainless steel support is 1.5 cm².

4. Centering Force and Support Systems

The straight section of the 10 T TF coil conductor is 4.26 m long with ampere-turn totaling 59.52×10^6 , the total centering force is 282×10^6 lbs., resulting in a radial compressive pressure of 8150 psi. If the rigidity of the coil form and the TF coil winding is not taken into consideration, and if 316 stainless steel is used, the circumferential design stress of a 17 cm thick support cylinder is 45,000 psi. The inner radius of the support cylinder is 0.68 m and the outer radius is 0.85 m. The TF coil is operated at 3°K so the support cylinder is at 3°K. The support cylinder is supported by a fiberglass cylinder.

Each TF coil is connected to its neighbors on either side by several cold supports. Since the poloidal field interacts with the TF coil, exerting a overturning out-of-plane load on the TF coil, some of the cold supports are cold shear plates resisting the twisting forces in a manner like a cross-bracing torque frame. Each TF coil has a torque bar connecting to the warm torque frame.

5. Coil Protection System

If any of the 12 coils carries unequal current, there is a large unbalanced force between adacent coils. For this reason, the series coil protection scheme is used in which each coil is connected in series with a dump resistor between each coil so that current in each coil will be equal at all times. The total energy for the 12 TF coils is 3100 MJ, so: that

the energy per coil is 258 MJ. With operational current of 10,000 amperes, the total inductance per coil is 61 H or 5.08 H per coil. If the dumping voltage of 1000 volt per TF is tolerable, and each dump resistor has resistance of 0.1 Ω , the dumping time constant of the TF coil system is 51 sec.

6. Cryostat Design and Cryostat Heat Leak

Each TF coil weighs approximately 35 tons. This gravitational load is supported by the support cylinder on the inside and by individual supports on the outside. The outer supports are made of Fiberglass G-10 laminate, loaded primarily in compression and transmitting the load through the vacuum jacket wall. The outer supports are fixed to the TF coil through a sliding joint to allow thermal contraction.

Each TF coil has a helium container made of stainless steel of 1.5 cm in thickness. The helium container conforms closely to the coil geometry with exception of 5 cm radial spacing reserved for 3°K condenser and gas liquid helium volume outside the winding. No such radial spacing is allowed, however, for the straight section coil form because the winding must transmit the centering force onto the helium can which in turn presses on the support cylinder.

The central region of the TF coil assembly, consisting of the inner part of the TF coil (straight section) and the support cylinder has a common vacuum jacket. There is a transition to individual vacuum jackets for the outer portions of the TF coil to provide the required access between coils. The outer portion of the vacuum jacket conforms with the coil form. It is estimated that the total helium temperature surface exposed to the 88°K shield is about 1000 m². Using 65 mW/m² as the radiation heat flux between the 88°K shield and liquid helium temperature vessel, the total radiation heat flux is estimated at 65 watts.

The gravitational support system must support 12 coils weighing 35 tons each and the support cylinder weighing 35 tons, for a total of 0.9×10^6 lbs. For fiberglass epoxy laminated with a design compressive stress of 30,000 psi, the total area required is 193 cm². If the length of support is 25 cm, the heat condition loss is about 10 watts.

7. Estimation of ac Loss in Coil Winding and Coil Form

For a simplified estimation of ac loss in the coil winding, the averaged cross section of NbTi per turn is approximately 0.1 cm^2 . This is a rough estimation based on a design averaged current density of NbTi at $14 \times 10^4 \text{ A/cm}^2$ at 3°K operation. The total amount of NbTi is estimated at 0.1 m^3 per TF coil. Assuming a 10μ filament diameter for the braided conductor strand, for a field swing of $0 \rightarrow 3 \text{ kG} \rightarrow 0$, we have filamentary loss per TF coil equal to 1030 J . To estimate the matrix loss per TF coil, it is assumed that the strand diameter of 0.5 mm is used in the braided cable and that a twist pitch of 4 mm can be achieved. The total composite volume per TF coil is estimated at 2 m^3 . The matrix loss for a field cycle consists of linear fall with $\dot{B} = 0.2 \text{ T/sec}$ and $B = 0.4 \text{ T}$ is 160 J per TF coil. The eddy current loss in the stainless steel support and the eddy current flowing across the strands within the braided cable have not been included in the loss estimation.

The superimposing field at the TF coil has both parallel and perpendicular field components. The ac field magnitude varies along the TF coil. For a worst space-averaged superimposing field, the transverse field dissipation for a complete cycle is estimated at 1600 J per TF coil. The parallel field loss is estimated at 1400 J per cycle per TF coil.

The summation of the ac loss in coil winding and the ac loss in coil form gives 4200 J per cycle per TF coil. That is, the power dissipation averaged over a cycle of 60 sec is about 70 watts per TF coil or a total of 840 watts for 12 TF coils .

8. The 3°K Refrigeration

The 10 T TF coil designed is based on the superconductor NbTi operated at 3°K to be economically feasible because the current carrying capacity of NbTi at 4.2°K decreases to a very low value at a 10 T magnetic field level. This is illustrated in Fig. IV-4. To date, refrigeration at temperatures lower than 4.2°K has been accomplished by vacuum pumping to maintain the required vapor pressure over a liquid helium bath. For small systems, this method is straight forward and has the virtue of adding

Figure IV-4. J-H curve for NbTi versus temperature.

only simple components to existing liquid helium facilities. As systems become larger, or the refrigeration capacity becomes larger, it is economically desirable to recover the cold vapor by adding a heat exchanger. However, as refrigeration capacity increases, the size of the heat exchanger becomes very large because of the necessity of maintaining the pressure drop in the heat exchanger at a low value. Furthermore, the vacuum pressure of a pumped system is more likely to introduce a serious problem of contamination because of leakage into the helium system in the recovery loop.

ANL has proposed a novel 3°K refrigeration scheme using super-critical dense helium gas as the working fluid. As shown in Fig. IV-5, high pressure helium is cooled to about 4.2°K by a counter flow heat exchanger and a liquid helium bath. This helium is then run through a low temperature exchanger and an expansion engine to reduce its temperature to about 2.2°K. This helium which remains supercritical is then used to condense helium gas in the closed TF coil form, creating a vacuum and thereby reducing the temperature of the liquid helium bath to about 3°K. The circulating helium, at a pressure of ~ 5 atm, then passes up the two low-pressure heat exchangers to the compressor suction side. The pressure and temperatures indicated in Fig. IV-5 represent tentative numerical values only. The pertinent operating temperature and pressure require detailed thermodynamic study on possible cycles, the effect of operating pressure, the pressure ratio and on the expander efficiency operating below 4.2°K.

As shown in Fig. IV-5, the 3°K condenser is placed within the outer leg of the D-shaped coil form so that condensation of gas bubbles will produce a difference between column weights of the inner leg and that of the outer leg. This difference plus the frictional drag force which exists between the upcoming gas and the liquid within the inner leg of the D-shaped coil will drive the liquid helium in circulation, resulting in greater heat transfer. To accommodate the volume of gas and the volume of the 3°K condenser, and in order to maintain the liquid level above coil windings, a 5 cm additional radial thickness of coil form is tentatively reserved for these purposes. Preliminary calculation shows that the size of the 3°K condenser for a 100 watts refrigeration capacity will be approximately 1 cm x 25 cm x 500 cm only. A total of 12 units of 100-watt 3°K refrigerators will be needed in the TETF TF Coil System.

Figure IV-5. Pure-tension TF coils, cryostat and a proposed 3°K refrigeration scheme.

C. Equilibrium Field Coil

1. Equilibrium Field Requirements

To provide the equilibrium for the plasma loop, the vertical field must have a strength equal to

$$B_{eq} = \frac{I_p}{R_o} \left[\ln \left(8 \frac{R_o}{a} \right) + \beta_\theta + \frac{\ell i - 3}{2} \right] \times 10^{-6} \text{ (kG)}$$

where β_θ is the ratio of plasma pressure to poloidal field pressure and is equal to $\Sigma nkT / (B_p^2 / 8\pi)$. The constant ℓi denotes the internal inductance of the plasma loop per unit length and it depends on the plasma current distribution over the plasma cross section. For uniform current density in the plasma, $\ell i = 0.5$. The I_p is the plasma current in amperes, R_o is the toroidal radius in meters, and a is the plasma radius in meters. Note that B_{eq} is the neutral equilibrium field while B_v in the following will denote the stable equilibrium field.

For stability against radial displacement, Greene et al¹ have shown that the vertical field component, B_v , must be smaller than the holding field B_{eq} for $R < R_o$ and B_v must be greater than B_{eq} for $R > R_o$, where R_o is the major radius of the plasma. That is, the condition

$$\left| \frac{\partial B_v}{\partial R} \right| < \left| \frac{\partial B_{eq}}{\partial R} \right|$$

is needed for stability against radial displacement.

For stability against vertical displacement, Yoshikawa³ has shown that the required condition is that the vertical field lines must be concave toward toroidal axis so that resulting force tends to center the plasma on the axis $z = 0$.

In the ANL - TETF design, we have $I_p = 1.4 \times 10^6$ A, $R_o = 3.0$ m, $a = 0.55$ m and $\beta_\theta = 3.4$. Therefore B_{eq} at plasma center is 2765 G.

2. Design Considerations for Equilibrium Field (EF) Coil

If the equilibrium coils are placed inside the TF coils, there are tremendous difficulties in coil assembly, disassembly, coil support, coil repair and maintenances, as well as the assembly, disassembly and maintenance of the blanket and shield and the first wall. To avoid these problems, the EF coils are located outside the TF coils.

The OH coils could induce large voltages in the EF coils, if the EF coils are not decoupled to the OH coils. The decoupling ampere-turns should be positioned so that they produce small anti-vertical fields and small superpositioning fields on the TF coils.

The EF coils were designed to produce as much volt-sec as possible for the plasma, thereby reducing the volt-sec requirement for the ohmic-heating coils.

3. Copper EF Coil Design

To design the water-cooled copper coil, the following conditions are used in the design calculations: (1) the EF coil is charged from 0 to 2.765 kG in 2 sec. It remains at 2.765 kG for 41 sec., then it is ~~discharged~~ back to zero field in 2 sec. After this, the EF coils remain at zero field for 15 sec during the pump-down time. So a complete cycle takes 60 sec., (2) the mean operating temperature of copper is 50°C. The resistivity of copper at 20°C is $1.7 \times 10^{-6} \Omega\text{-cm}$ and a temperature coefficient of $3.9 \times 10^{-3}/^\circ\text{C}$ is used to compute the resistivity of copper at 50°C, which is $1.9 \times 10^{-6} \Omega\text{-cm}$. A practical inlet water temperature of 5°C was selected. A water temperature gradient of 90°C was chosen to keep the coil temperature well within a range where a conventional epoxy-impregnated structure is ~~usable~~ (3) The water circuit design was assumed to be flexible so that average ~~curr~~ density over conductor cross section can be varied from 750 A/cm² to about 1500 A/cm². This low current density is necessary to reduce the I²R loss of the water-cooled copper coil.

The instantaneous I²R loss for water-cooled copper coil is given by

$$P_{ins} = J_c^2(t) \rho V_c$$

where $J_c(t)$ is the conductor current density, ρ is the resistivity of copper and V_c is the conductor volume.

Based on the EF coil design considerations and the EF requirements, the EF coil configuration was designed as shown in Fig. IV-1.

To produce a required equilibrium field of 2.765 kG at plasma center, the required ampere-meters are 153×10^6 . If the conductor current density is 1500 A/cm^2 , the total conductor volume is 10.2 m^3 . The instantaneous I^2R loss is 43 MW neglecting the eddy current loss in the coil. The cycle-time-averaged power dissipation is 30 MW. The cost of 10.2 m^3 copper conductor only amounts to about \$0.8 million dollars. Yet it is highly desirable to keep the power supply requirement at a reasonably low level so that expensive energy storage and power supply can be avoided in the TETF facility. For this reason, the designed copper current density is chosen to be 750 A/cm^2 with total copper volume equal to 20.4 m^3 . The instantaneous power is 21.5 MW and the time averaged power dissipation is 15 MW. The coil cross sections corresponding to this low current density design is shown in Fig. IV-1a.

To keep the voltage of coil and the power supply to order of 10 kV or lower, the operational current of 30 kA is chosen. The self inductance is equal to 0.15 H. The total dc resistance is 0.024Ω and the total coil impedance is 0.237Ω . Therefore the design voltage required to drive the EF coil is about 5 kV.

The total ampere turns required is 9.6×10^6 , with half of these ampere-turns serving to magnetically decouple the equilibrium coils from the OH coils. These antimutual inductance coils are coils No. 5, 6, 7, and 8. They are positioned to generate negligible vertical fields in the negative z direction everywhere on the $z = 0$ axis, thus enabling coils No. 1, 2, 3 and 4 to produce as much useful vertical field as possible. Furthermore, this coil arrangement generates the most volt-sec for a given equilibrium field. The total volt-sec contributions from the EF coil is 7.8 volt-sec.

Several additional coils may be required to form a hexapole null in the torus to localize the region of plasma current startup at the beginning of each pulse. The filling gas is broken down in this region

by means of a microwave pulse at the electron cyclotron resonant frequency. The hexapole field acts as an expanding magnetic limiter during the early stage of current ramp-up. Once the plasma current reaches about 400 kA, these coils are discharged

4. Superconducting EF Coil Design

As a design option for EF Coil system, a superconducting EF coil design was examined. If the EF Coil is superconducting, the ac pulsing loss will be much smaller than I^2R loss of water-cooled copper coil. The design current density could be as high as 3000 A/cm^2 and cryogenic stability can still be achieved if the composite strand diameter is about 0.6 mm. Hence the winding cross section of EF coil will be 4 times smaller than that of copper EF coil as shown in Fig. IV-1a.

The 2 sec fast pulsing suggests that the filaments must be twisted in a short pitch. The 30 kA conductor will require $\sim 10^6$ fine filaments. It is impossible to twist 10^6 filaments in a short pitch. This suggests that the conductor can not be made of one strand. Cabling conductor made of many strands with full transposition within the cable is necessary. Cable conductors made of many fully transposed strands often result in a poor packing factor, poor mechanical rigidity and possible shorts between strands. Hence, a composite with a large number of filaments was chosen. Such a composite will have a large diameter and thus a 3-component composite with high resistive barriers of cupro-nickel between filaments is necessary to reduce the matrix loss. On the other hand, the EF coil has a stored energy requiring full stabilization. This means that the composite diameter can not be too large in order to ensure a large surface-to-volume ratio for cooling necessary for full stabilization. A suitable specific superconducting composite specification is listed in Table IV-3. Two 30-kA superconducting cable designs using the specific superconducting strand are shown in Fig. IV-6.

The total length of the 61.7 A composite is 2.48×10^6 meters. Therefore the total composite volume is 0.7 m^3 . For a 5μ filament diameter and field excursion of $0 \rightarrow 30 \text{ KG} \rightarrow 0$ in 4 sec, the estimated hysteresis loss is 2300 J per cycle and the matrix loss is 1440 J. The self-field loss is negligible. For a 60 sec cycle time, the average power dissipation is about

TABLE IV-3. Superconducting Strand Composite Properties

NbTi critical-current density	$1.9 \times 10^5 \text{ A/cm}^2$ at 4.2°K, B.7
Composite operational current	61.7 A
Composite critical current	123.4 A
Composite diameter	0.6 mm
Composite NbTi packing factor	0.23
Composite composition NbTi:Cu:Cupro-nickel	0.23:0.57:0.20
Filament diameter	5 μm
Twist pitch	6 mm
Number of filaments in composite	3307
Composite operational current density	$2.185 \times 10^4 \text{ A/cm}^2$
Composite critical current density	$4.37 \times 10^4 \text{ A/cm}^2$
Composite matrix effective resistivity	$10^{-6} \Omega\text{-cm}$

62 W at 4.2°K or 31 kW at 300°K. These loss calculations do not include the eddy current dissipations in the coil supporting structures and in the helium vessel. If the conductor cable is enclosed in a stainless steel tubing with a 3 cm diameter and a wall thickness of 1.4 mm, the energy dissipation in the tube will exceed the ac pulsing loss in the NbTi and the matrix.

Preliminary comparisons made between the water-cooled copper EF coil and the superconducting EF coil shows a substantial power saving can be made if the coil is superconducting. Unfortunately the technology of ac superconducting magnets is not yet ready for such large stored energy and fast pulsed EF coil and therefore extensive R and D will be needed if the EF coil is superconducting. On the other hand, a power consumption of the order of 50 MW is required with the copper EF coil.

D. Ohmic Heating Coils

1. The Volt-sec Requirement for the OH Transformer

During the startup period, when the electron temperature is low, the plasma loop is more of a resistive than an inductive load. When electron temperature ~ 0.5 keV, the plasma loop becomes mostly purely inductive with a very small resistive component. During the entire startup period, it is estimated that the resistive volt-sec will be 2.4 volt-sec while the inductive volt-sec will be 9.6 for the TETF non-circular plasma having major radius of 3.0 m and height-to-width ratio, κ , equal to 1.3. So the total volt-sec during startup is 12. During the burn cycle, an additional core flux of 5 volt-sec, is reserved to maintain the plasma current.

The EF coil is designed to produce as much as 7.8 volt-sec during the startup period. Therefore, the OH coil is required to produce 4.2 volt-sec in the startup period and 5 volt-sec during the burn phase.

2. Water-cooled Copper OH Coil

Because the TF coil is superconducting, the OH coil must be placed outside the TF coils in such a way that the superimposing field on the TF coils is minimized. The OH coil windings must also

be arranged so that the field produced by the OH coils is nearly zero in the entire plasma region, and the plasma can be initially ionized in a stable equilibrium. To achieve these two purposes as well as the required flux linkage for volt-sec requirements, the OH coil configurations were designed as shown in Fig. IV-1a.

An effort was made to keep the total power requirement of the TETF to about the 100 MW level, so that no major cost in the energy storage and power supply is needed. During the startup period, 21 MW of resistive power and 34 MW of inductive power are required for the EF coils. Therefore, the charging scheme of the OH coils must be such that it evens out the power demand over the entire cycle. For this consideration, the OH coils are required to produce 4.2 volt-sec during the startup period and 5 volt-sec during the burn phase. Since the radius of the flux core is 0.5 m, the required central field is -5.35 T before startup and + 6.37 T at the end of the burn phase. The operational current density will be 2128 A/cm² before the startup and 2534 A/cm² at the end of burn phase. The conductor required is 270×10^6 ampere-meters before the startup and 321×10^6 ampere-meters at the end of the burn phase. The total conductor volume is 6.3 m³, so that the instantaneous I^2R loss is 54 MW at $t = 0$ and 77 MW at the end of the burn cycle. The inductive stored energy in the OH coil self-field system will be 82 MJ for the 5.35 T central field and 116 MJ for the 6.37 T central field.

If the operational current of 30 kA is chosen for the 5.35 T central field, the self inductance of the OH coil is 0.182 H. The total dc resistance is 0.06 Ω and the total coil impedance is 0.292 Ω . Therefore, the design voltage required to drive the OH coil is 5000 volts.

3. Superconducting OH Coil Design

As a design option for the OH coil system, the superconducting OH coil design was examined. The high current fast pulsing superconducting OH coil requires a similar cable design as shown in Fig. IV-6. With the 6 mm diameter strand, rough estimations shows that the cryogenic stability can be maintained even if a conductor current density as high as 3000 A/cm² is used. However, if the OH coil is superconducting, the flux core size will be 0.56 m in radius because the OH coil is placed adjacent to the support cylinder of the TF coil. Consequently, the central field before the startup

is -4.28 T instead of -5.35 T and the central fields +6.37 is replaced by 5.1 T. This reduces the stored energy of the OH coil system. In fact, because the ac pulsing loss of superconducting magnets is very small, it is not necessary to have a charging scheme described in the previous copper OH coil section. The sensible charging scheme for the superconducting OH coil is the one in which the ampere-turns are minimized. That is, the negative swing central field will be equal to the positive swing central field and both are equal to 4.69 T, a desirable field range for the superconducting magnet using NbTi as the superconductor.

To estimate the ac pulsing loss and compare this with the I^2R loss of the copper magnet, the calculation of conductor length is based on the winding dimensions of copper OH coil as shown in Fig. IV-1a. Using the 61.7 A composite strand of diameter 0.6 mm as described in Table IV-3, the total length of the composite strand will be 5.2×10^6 m where the conductor length for the central field of 6.37 T is used. The total composite volume is 1.47 m^3 . Using the 5μ as the filament diameter and a field excursion of $-6.37 \text{ T} \rightarrow +6.37 \text{ T} \rightarrow -6.37 \text{ T}$, the hysteresis loss over a complete fusion cycle is equal to 29930 J. The matrix loss for a field swing of $-6.37 \text{ T} \rightarrow +6.37 \text{ T} \rightarrow -6.37 \text{ T}$ with $\dot{B} = 6.37 \text{ T/sec}$ and filament twisting pitch of 6 mm is equal to 11500 J. Therefore, the total ac loss from the conductor over a complete fusion cycle is 41430 J. For a representative cycle time of 60 sec, the average power dissipation at 4.2°K is 690 W. The required refrigeration at 300°K is 345 kW. It is clear that superconducting OH coil will have less power loss than the copper OH coil. On the other hand, the superconducting OH coil will require more R and D than the copper OH coil.

E. Eddy Currents in the Shield

The time-varying magnetic fields from the equilibrium coils and from the plasma currents cause eddy currents in the shield. The magnetic fields produced by these eddy currents may disturb the plasma, in a manner which depends on the magnitude, homogeneity, and time delay of these induced fields. In addition, there may be unacceptably large forces, torques, stresses, or energy dissipation in the shield.

The eddy currents and their effects are diminished by breaking the shield into pieces, electrically insulated from each other. The shield, as described in Chapter VIII, consists of twelve units, each positioned between two neighboring TF coils. Each unit, in turn, consists of nine blocks. For computational purposes, these blocks were taken as rectangular stainless steel bricks, aligned parallel to the coordinate axis. The eddy current distribution in the bricks and the field resulting from these currents were calculated by a method described elsewhere.² The peak induced field in the plasma was found to be 0.34% of the final equilibrium field, or 9 gauss, which is judged to be acceptably low. In so far as the average electrical conductivity of the blocks is actually less than that of stainless steel, the induced field will be reduced proportionally.

The eddy currents produce torques in the shield. The torque, acting on each of the largest blocks due to the 4.0 T field from the toroidal field coils interacting with the eddy currents, is 46000 ft lb. The blocks must be restrained to resist this torque.

F. The TETF as a Magnet System Demonstration Facility

The TETF is a prototypical demonstration of the superconducting TF coil system that must be developed for the EPR and subsequent tokamak reactors. The TETF magnet bore is about one-half that of the EPR, and both have a pure-tension D shape. Conductor design, coil winding, coil protection, cryostat and 3°K refrigeration system for TETF are prototypical for EPR.

The superconducting EF coil and OH coil options for TETF would be prototypical of pulsed poloidal coils for EPR and future reactors. However, the copper coils, which are currently the primary option, would probably represent a technology culmination in the tokamak program, with no further extrapolations in future reactors.

References

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2. CTR Program, "CTR Quarterly Progress Report, July-September, 1975," Argonne National Laboratory, ANL/CTR-75-3, pp. 54-58 (November, 1975).
3. S. Yoshikawa, "Application of the Virial Theorem to Equilibria of Toroidal Plasmas," *The Physics of Fluids* 7, p. 278 (1964).

V. NEUTRAL BEAM INJECTORS

The neutral beam injector system for TETF consists of eight injectors, with two beam lines per injector. Four of the injectors (eight beam lines) provide 15 MW of 60 keV D^0 beam; the other four injectors provide 15 MW of 90 keV T^0 beam. Each beam line consists of an ion source and accelerator followed by a double focusing bending magnet to separate molecular ions from atomic ions, a neutralizer for the atomic ions and a second bending magnet to remove ions from the beam emerging from the neutralizer. Direct energy converters are used to recover energy from the ions removed by the first and second separators. The two neutral beams from each injector converge toward an 0.55 m wide, 0.8 m high aperture in the first wall of the torus at angles of 6° above and 6° below the horizontal median plane of the torus.

A. Ion Sources

The preliminary design for the TETF neutral beam injectors is based on the use of state-of-the-art CTR ion sources. The final design will take advantage of any improvements that occur prior to the design freeze date.

The assumed ion beam composition is typical of present-day LBL sources.¹ Water-cooled multiaperture grids capable of dc operation will be of the type developed at ORNL.² Circular apertures will be used in preference to slots so that the horizontal and vertical transverse beam emittances will be identical; this will simplify the design of the double focusing magnet used to remove molecular ions from the beam entering the neutralizer and to focus the atomic and neutral beams.

The design characteristics of the ion sources are given in Table V-1. It is assumed that the extraction electrode spacings and voltages are the same in both kinds of sources; therefore, current densities are proportional to $M^{-1/2}$, where M is the effective mass number of the beam.

Ion beam current requirements are given by

$$I_+ = \frac{P_B / V_+}{\epsilon_R \Gamma_1 f_{10} e^{-\pi_+ \sigma_{10}} e^{-\pi_0 \sigma_{01}}} \cdot A \quad (1)$$

TABLE V-1. Ion Source Design Characteristics

	<u>D^o</u>	<u>T^o</u>
D ⁺ (T ⁺) fraction of I ₊ , Γ ₁	0.75	0.75
D ₂ ⁺ (T ₂ ⁺) fraction of I ₊ , Γ ₂	0.18	0.18
D ₃ ⁺ (T ₃ ⁺) fraction of I ₊ , Γ ₃	0.07	0.07
Plasma Ion Emission Current Density, J _o (A/cm ²)	0.300	0.245
Type of Grid	Multiperture	Multiperture
Grid Transparency, τ _g	0.45	0.45
Ion Beam Current Density, J ₊ = τ _g J _o (A/cm ²)	0.135	0.110
Gas Efficiency, η _g	0.45	0.45
Pulse Length	dc	dc

In this expression, P_B is the neutral beam power; V_+ is the ion beam voltage; ϵ_R is the reactor capture efficiency; Γ_1 is the atomic fraction of I_+ ; f_{10} is the power efficiency for $D^+ \rightarrow D^0$ or $T^+ \rightarrow T^0$; $e^{-\pi+\sigma_{10}}$ is the fraction of the $D^+(T^+)$ beam which reaches the neutralizer, the remainder $(1 - e^{-\pi+\sigma_{10}})$ being lost due to premature neutralization along a curved flight path while traversing a gas target of π_+ atoms/cm²; $e^{-\pi_0\sigma_{01}}$ is the fraction of the $D^0(T^0)$ beam, which reaches the reactor plasma, the remainder being lost due to re-ionization in a gas target of π_0 atoms/cm²; σ_{10} is the electron capture cross section (cm²/atom); and σ_{01} is the electron loss cross section (cm²/atom).

In terms of average pressure $\bar{p}$ (Torr), path length ℓ (m) and gas temperature T (°K), the gas target density is given by

$$\pi = 1.94 \times 10^{21} \bar{p}\ell/T \text{ atoms/cm}^2 \quad (2)$$

B. Atomic Ion and Neutral Beam Transport

A preliminary $D^0(T^0)$ beam line design* is shown in Fig. V-1. M-1 is a double focusing magnet for D^+ or T^+ with field index $n = -(R/B)dB/dR = 1/2$, bending radius $R_0 = 1.5$ m and bending angle $\theta = 45^\circ$; for a 60 keV D^+ beam, the gap field $B_0 = 333$ gauss. Field variation with R is given by $B = B_0 (R/R_0)^{-1/2}$.

The ion beam emerging from the accelerator is assumed to be cylindrical and axially symmetric. Beam transport calculations for motions of atomic ions and neutrals in the transverse phase spaces $(x, x'; y, y')$ have been made for several values of emittance $\epsilon = \epsilon_x = \epsilon_y$. ϵ is determined by the initial beam size $2r_{0 \text{ max}} = 2x_{0 \text{ max}} = 2y_{0 \text{ max}}$ and the maximum initial beam-let half-angle divergence $r'_{0 \text{ max}} = x'_{0 \text{ max}} = y'_{0 \text{ max}}$. The area of the transverse phase space ellipse at the exit of the accelerator, where the initial conditions occur, is given by

$$\gamma_0 r_0^2 + \beta_0 (r'_0)^2 = \epsilon = \frac{\text{area}}{\pi} = (r_{0 \text{ max}})(r'_{0 \text{ max}}), \quad (3)$$

which gives the relationship between ϵ , $r_{0 \text{ max}}$ and $r'_{0 \text{ max}}$.

* We are indebted to T. K. Khoe of the Accelerator Research Facilities Division, ANL, for the design of the separator and the beam transport calculations.

Figure V-1. Preliminary $D^0(T^0)$ beam line design. R_0 is the bending radius, θ is the bending angle in M-1.

Results of some transport calculations are given in Table V-2 for D^+ and D^0 . The results apply to T^+ and T^0 beams with the same initial beam sizes and divergences as well. However, the T^+ beams will be smaller than the D^+ beams and will be considered separately for design purposes.

C. Molecular Beam Transport

For the purposes of this discussion, the plane defined by the x-axis and the beam axis will be called the horizontal plane (plane of the paper in Fig. V-1) and the plane defined by the y-axis and the beam axis will be called the vertical plane. For the molecular ion beams, the bending angles are not equal to the magnet angle. Therefore, the molecular beams experience vertical defocusing and horizontal focusing at the downstream edge of the magnet. Results of transport calculations for D_2^+ and D_3^+ are given in Table V-3, where S is the distance from the downstream magnet edge; initial conditions are those of Table V-2.

D. Injector Power Flow and Overall Efficiency

The injector power flow diagram is shown in Fig. V-2. Ion beam from the accelerator enters a magnetic separator, M-1, which removes molecular ions from the beam before it enters the neutralizer. The molecular ion beams, with some attenuation due to charge-changing collisions and dissociation, go to a direct energy converter where their energy is recovered with an efficiency $\eta_1 \approx 0.85$. A detailed design of a direct converter for TETF has not yet been done but preliminary considerations suggest that a space charge controlled converter³ would be more suitable than an immersed grid converter.⁴

As noted above, in the discussion of Eq. 1, there is attenuation of the atomic ion beam due to premature neutralization along its curved flight path and attenuation of the neutral beam due to reionization.

In a field-free region, loss of D^+ or T^+ , i.e., growth of D^0 or T^0 , is given by

$$F_0 = F_{0\infty} [1 - e^{-\pi(\sigma_{01} + \sigma_{10})}] \quad (4)$$

where $F_{0\infty}$ is the equilibrium (thick target) fraction of D^0 or T^0 . Similarly, loss of D^0 or T^0 (growth of D^+ or T^+ in a beam that is initially neutral) is given by

TABLE V-2. Results of D^+ , D^0 Beam Transport Calculations for the Beam Line of Fig. V-1.

	<u>Case I</u>	<u>Case II</u>	<u>Case III</u>
Initial beam radius, $r_{0 \text{ max}}$ (m)	± 0.15	± 0.15	± 0.13
Initial beamlet divergence, $r'_{0 \text{ max}}$ (rad)	± 0.020	± 0.030	± 0.030
Transverse emittance ϵ (m-rad)	3.0×10^{-3}	4.5×10^{-3}	3.9×10^{-3}
Maximum beam size in magnet (m)	0.301	0.301	0.261
Beam size at neutralizer entrance (m)	0.241	0.252	0.223
Beam size at neutralizer exit (m)	0.158	0.236	0.236
Distance, magnet exit to beam waist (m)	2.65	1.965	2.046
Beam size at waist (m)	0.149	0.205	0.199
Beam size at first wall of torus (m)	0.404	0.512	0.491

TABLE V-3. Results of D_2^+ , D_3^+ Beam Transport Calculations for the Beam Line of Fig. V-1.

	<u>Case I</u>		<u>Case II</u>	
	<u>S = 2 m</u>	<u>S = 4 m</u>	<u>S = 2 m</u>	<u>S = 4 m</u>
D_2^+ horizontal beam size (m)	0.182	0.174	0.223	0.261
D_2^+ vertical beam size (m)	0.294	0.328	0.333	0.412
D_3^+ horizontal beam size (m)	0.196	0.178	0.236	0.264
D_3^+ vertical beam size (m)	0.341	0.407	0.379	0.485

Figure V-2. TERF injector power flow diagram. π is a gas target thickness (atoms/cm²) for charge-changing collisions or dissociation. η_1 and η_2 are electrical energy recovery efficiencies (≈ 0.85). $f_{10} = 0.95 f_{0\infty}$ = power efficiency for $D^+ \rightarrow D^0$ or $H^+ \rightarrow H^0$.

$$F_1 = F_{1\infty} [1 - e^{-\pi(\sigma_{01} + \sigma_{10})}] \quad (5)$$

In magnetic field regions, particles that undergo charge changing collisions are lost from the beam. In such regions, $F_{0\infty} \rightarrow 1.0$ in Eq. 4 and $F_{1\infty} \rightarrow 1.0$ in Eq. 5. Eq. 4 must then be replaced by

$$F_0 = 1 - e^{-\pi \sigma_{10}} \quad (4-A)$$

and Eq. 5 by

$$F_1 = 1 - e^{-\pi \sigma_{01}} \quad (5-A)$$

For thin targets, i.e., everywhere in the neutral beam injector except the neutralizer, Eqs. 4 and 4-A may be used interchangeably, without regard to the presence or absence of magnetic field; Eqs. 5 and 5-A also yield virtually identical results. Use of Eqs. 4-A and 5-A for both field and field-free regions in Fig. V-2 has resulted in a flow diagram far less complicated than it would otherwise have been.

A number of cross sections for charge changing collisions and dissociation appear in Fig. V-2. Since the velocities of 30 keV hydrogen particles, 60 keV deuterium particles and 90 keV tritium particles with the same number of nucleons per particle are equal, their cross sections are also equal. Thus, the summations in Fig. V-2 may be written out for 30 keV hydrogen particles and then used for the corresponding 60 keV deuterium particles and 90 keV tritium particles:

$$\Sigma(\sigma_{\text{loss}})_2 = \sigma_{H_2} + 1/2 (\sigma_H + \sigma_{H^+}) \quad (6)$$

$$\Sigma(\sigma_{\text{loss}})_3 = 1/3 (\sigma_H + \sigma_{H^+}) + 2/3 (\sigma_{H_2} + \sigma_{H_2^+}) \quad (7)$$

Values for all of the cross sections, measured for 30 keV hydrogen particles, are given in Table V-4.

TABLE V-4. Cross Sections for Charge Changing Collisions and Dissociation for 30 keV Hydrogen, 60 keV Deuterium and 90 keV Tritium Particles

<u>Particle</u>	<u>Cross Sections</u>	<u>Reference</u>
30 keV H^+ or H^0 or equivalent ($cm^2/atom$)	$\left\{ \begin{array}{l} \sigma_{10} = 19.3 \times 10^{-17} \\ \sigma_{01} = 6.5 \times 10^{-17} \end{array} \right.$	 5 5
30 keV H_2^+ or equivalent ($cm^2/molecule$)	$\left\{ \begin{array}{l} \sigma_H = 7.8 \times 10^{-16} \\ \sigma_{H_2} = 2.65 \times 10^{-16} \\ \sigma_{H_2^+} = 1.8 \times 10^{-16} \end{array} \right.$	 6 6 6
30 keV H_3^+ or equivalent ($cm^2/molecule$)	$\left\{ \begin{array}{l} \sigma_H = 9.51 \times 10^{-16} \\ \sigma_{H_2} = 4.36 \times 10^{-16} \\ \sigma_{H^+} = 1.77 \times 10^{-16} \\ \sigma_{H_2^+} = 1.17 \times 10^{-16} \end{array} \right.$	 6 6 6 6

The values of σ_{10} and σ_{01} in Table V-4 are averages of several values compiled by S. K. Allison and M. Garcia-Munoz.⁵ Values of $F_{0\infty}$ and $F_{1\infty}$ are also given in reference 5; for 30 keV H^+ , $F_{0\infty} = 0.71$ and $F_{1\infty} = 0.014$.

The gas flow diagram for TEF is shown in Fig. V-3. The various gas loads are defined in Table V-5.

The fractional beam losses $1 - e^{-\pi+\sigma_{10}}$ and $1 - e^{-\pi+\sigma_{01}}$ depend on the pressure distributions in Chambers I and II of Fig. V-3. The pumping speed in the region of M-1 will be conductance-limited to a value that will not permit arbitrary reductions in pressure and beam loss for a given gas load.

All of the gas from the accelerator and the neutralizer enters a volume bounded by the exit plane of the accelerator, the planes of the polepieces and the plane defined by the entrance face of the neutralizer. The gas leaves this volume through the unobscured fraction of the bounding surface, which is maximized by using an open "C" type bending magnet rather than a "window frame" type. The area of this fraction is estimated to be 2.5 m^2 for the beam line design of Fig. V-1. For deuterium, the room temperature conductance of this area considered as an aperture is 7.8×10^5 liter/sec. For a Chamber I pumping speed of 1×10^6 liter/sec, it follows that the effective, conductance-limited pumping speed S in the M-1 region will be $\sim 4.4 \times 10^5$ liter/sec. Thus, the average pressure p_+ along the curved D^+ flight path will be

$$p_+ = (1/S) [Q_{\text{out}} + 1/2 Q_N] = 2.3 \times 10^{-6} (Q_{\text{out}} + 1/2 Q_N) \text{ Torr} \quad (8)$$

The neutralizer target thickness for a power efficiency $f_{10} = 0.95 F_{0\infty}$ is 5.80×10^{-6} Torr-meter/ $^\circ\text{K}$. For room temperature D_2 gas and a neutralizer length of 3 m, $\bar{p}_N = 5.67 \times 10^{-4}$ Torr if $p_{\text{exit}} = 0.1 p_{\text{center}}$. For a neutralizer radius of 13 cm it follows that $1/2 Q_N = 3.5$ Torr-liter/sec.

Substituting known values of P_N (15 MW), V_+ (0.06 MV) and r_1 (0.75) into Eq. 1 and assuming $c_R = 0.95$, $f_{10} = 0.68$, $e^{-\pi+\sigma_{10}} = 0.94$ and $e^{-\pi+\sigma_{01}} = 0.98$, we find $I_+ = 560$ A. For a single D^0 beam line, $I_+ = 560/8$ and Table V-5 gives $Q_{\text{out}} = 10.7$ Torr-liter/sec. Thus, $Q_{\text{out}} + 1/2 Q_N = 14.2$ Torr-liter/sec per beam line and Eq. 9 gives $\bar{p}_+ = 3.3 \times 10^{-5}$ Torr, which is consistent with $e^{-\pi+\sigma_{10}} = 0.94$.

TABLE V-5. Gas Loads to Pump Chambers I, II and the Torus

<u>Nature of Load</u>	<u>Amount of Load (Torr-liter/sec)</u>	<u>Pump Chamber</u>
Gas flow into source	$Q_{in} = 0.095 (I_+/n_g) \frac{3}{1} \Gamma_1$	-
Gas flow out of source	$Q_{out} = (1 - \eta_g) Q_{in}$	I
Premature neutralization of $D^+(T^+)$	$Q_{10} = 0.095 \Gamma_1 I_+ (1 - e^{-\tau + \sigma_{10}})$	I
Recombination of $D_2^+(T_2^+)$ on collecting surfaces	$Q_2 = 0.095 (I_+/n_g) \times 2 \Gamma_2$	I
Recombination of $D_3^+(T_3^+)$ on collecting surfaces	$Q_3 = 0.095 (I_+/n_g) \times 3 \Gamma_3$	I
Gas load to Chambers I and II from neutralizer ^a	$1/2 Q_N = 1/2 [4 \times 30.48 (a^3 k) (T/M)^{1/2}] (P_{center} - P_{exit})$	I, II
Recombination of $D^+(T^+)$ on collecting surfaces	$Q_{10}^+ = 0.095 \Gamma_1 I_+ [(1 - 0.95 F_{0m} - F_{1m}) e^{-\tau + \sigma_{10}}]$	II
Reionization of $D^0(T^0)$	$Q_{01} = 0.095 \Gamma_1 I_+ [(1 - e^{-\tau + \sigma_{01}}) f_{10} e^{-\tau + \sigma_{10}}]$	II
Recombination of $D^-(T^-)$ on collectors	$Q_- = 0.095 \Gamma_1 I_+ F_{1m} e^{-\tau + \sigma_{10}}$	II
Neutral beam	$Q_0 = 0.095 F_B / V + c_R$	Torus
Slow gas flux to torus per injector (2 beam lines) ^b	$Q_{D_2(T_2)} = C_{duct} (P_{injector} - P_{torus})$	Torus

a. Neutralizer radius a and length l measured in cm.

b. For a rectangular duct of cross sectional area A , length l and perimeter H (dimensions in cm) the conductance is given by⁷

$$C_{duct} = \frac{3.638 A (T/M)^{1/2}}{1 + 3/16 Hl/A} \text{ liter/sec}$$

Figure V-3. Gas flow diagram for TET-2. The various gas loads are defined in Table V-5.

From Table V-5 we find that the total gas load per beam in Chamber I is $1/2 Q_I = 22.9$ Torr-liter/sec. This gives an average pressure $\bar{P}_I = 2.3 \times 10^{-5}$ Torr in Chamber I for the design pumping speed of 1×10^{-6} liter/sec per beam line.

From Table V-5 we find that the total gas load per beam in Chamber II is $1/2 Q_{II} = 5.1 - 1/2 Q_{D_2}$ Torr-liter/sec, where Q_{D_2} is the slow gas flux to the torus. The neutral beam duct between the injector and the torus is 2.25 m long and 0.55 m wide; it tapers from a height of 0.9 m at the injector end to a height of 0.8 m at the first wall. Using the equation given in Table V-5 for the conductance of a rectangular duct, $Q_{D_2} = 41300 \Delta P$ and $Q_{II} = 10.2 - 41300 \Delta P$ Torr-liter/sec. If we assume $\bar{P}_{II} \sim P_{\text{torus}} = 1 \times 10^{-5}$ Torr, $Q_{II} \approx 10.2$ Torr-liter/sec; the pumping speed required to maintain a pressure of 1×10^{-5} Torr in Chamber II is 1.0×10^6 liter/sec.

It was assumed above that the neutral injector duct has a width of 0.55 m and a height of 0.8 m at the first wall. Table V-2 shows that a rectangular cross section of $0.55 \times 0.55 \text{ m}^2$ would be adequate. The consequences of reducing the height to 0.55 m are being considered.

Molecular ions traverse a path of ~ 1.5 m with an average pressure of 3.2×10^{-5} Torr, or 4.8×10^{-5} Torr meter of gas, in the conductance-limited region of the M-1 magnet. If they travel an additional 1.5 meters, to reach the direct converter, in a region with the average chamber pressure of 2.3×10^{-5} Torr, or 3.45×10^{-5} Torr meters of gas, the total gas line density is 8.3×10^{-5} Torr meter. Using the cross sections of Table V-3, we find that the fractional losses of D_2^+ and D_3^+ are identical and equal to 0.18. The total power recovered (see Fig. V-2) is thus

$$P_R = 0.82 \eta_1 (\Gamma_2 + \Gamma_3) P_+ + \eta_2 (1 - 0.68 - 0.014) \times 0.94 \Gamma_1 P_+ \text{ MW}$$

where numerical design values for f_{10} , F_{10} and $e^{-\pi + \sigma} 10$ have been used. Using the Γ values of Table V-1, this reduces to $P_R/P_+ = 0.22 (\eta_1 + \eta_2)$. The overall neutral injector power efficiency is given by

$$\epsilon_{NI} = \frac{P_R/P_+}{k - P_R/P_+} \quad (9)$$

Using the design values of P_B (15 MW) and P_+ (33.6 MW) and assuming $k = 1.1$, $n_1 = n_2 = 0.80$ we find $c_{NI} = 0.59$. The more conservative set $k = 1.1$, $n_1 = n_2 = 0.70$ gives $c_{NI} = 0.56$.

A detailed design of the second separator has not yet been done. Use of an in-line converter has been considered and tentatively rejected because the current density of the D^+ beam emerging from the neutralizer ($\sim 0.033 \text{ A/cm}^2$) is too high for efficient energy recovery (see Fig. V-2 and Eq. 18 of reference 3). A bending magnet (M-2) will, therefore, be used to spread the beam out as is done with the molecular beams in M-1. A larger bending angle will be used in the M-2 separator to make more efficient use of space.

The neutral beam injector design is summarized in Table V-6.

E. Layout of Injector Systems

The TETF has eight neutral beam injection ports as can be seen in Figs. XI-1 and XI-2. Four pairs of beams inject clockwise and four inject counter-clockwise in a symmetrical pattern. Each port is fed by a dual beam injector unit whose beams spread 6° above and 6° below the horizontal reactor centerline converging into a rectangular aperture .55 m wide by .8 m high at the vacuum vessel. The horizontal beam line is tangent to the poloidal plasma at a point half way on its inner centerline, $\sim 2.75 \text{ m}$ radius. A beam dump will be provided on the outer vacuum wall in line with the beam entry.

Reactor portions of the neutral beam port extending from the vacuum vessel to the disconnect point at the edge of the outer magnet coils include the following five components. A stainless steel rectangular duct is provided to channel the beam through the shield for a length of $\sim 2 \text{ m}$. The duct is water cooled with coils on the outer surface. Special shield blocks are provided in the reactor magnet shield to compensate for the neutral beam volume lost to the shield as well as the rearrangement of shield block structures required for assembly and support. A special shield is provided on the duct beyond the reactor shield.

A quick closing full port valve is provided at the outer edge of the shield to accommodate any pulsing mode of operation anticipated that might require isolation of the neutral beam injector. Beyond this valve is

TABLE V-6. Neutral Beam Injector Design Summary

	D^0	T^0
Neutral beam power, P_B (MW)	15	15
Neutral beam energy, W_B (keV)	60	90
Neutral beam current, I_0 (equiv. A)	250	167
Equilibrium neutral fraction, $F_{0\infty}$	0.71	0.71
Equilibrium $D^+(T^+)$ fraction, $F_{1\infty}$	0.014	0.014
Power efficiency for $D^+ \rightarrow D^0$, $T^+ \rightarrow T^0$, $f_{10} = 0.95 F_{0\infty}$	0.68	0.68
Neutralizer target thickness at 293°K for $f_{10} = 0.95 F_{0\infty}$, n_N (atoms/cm ²)	1.13×10^{16}	1.13×10^{16}
Neutralizer gas line density at 293°K (Torr meter)	1.7×10^{-3}	1.7×10^{-3}
D^+ , T^+ fractional loss, $1 - e^{-\pi + \sigma_{10}}$	0.06	0.06
D^0 , T^0 fractional loss, $1 - e^{-\pi 0^{\sigma} 01}$	0.02	0.02
D_2^+ , T_2^+ fractional loss, $1 - e^{-\pi_2 \Sigma(\sigma_{loss})_2}$	0.18	0.18
D_3^+ , T_3^+ fractional loss, $1 - e^{-\pi_3 \Sigma(\sigma_{loss})_3}$	0.18	0.18
Neutral beam capture efficiency, c_R	0.95	0.95
Ion beam composition (F_1 , F_2 , F_3)	(0.75, 0.18, 0.07) (0.75, 0.18, 0.07)	
Ion beam voltage, V_+ (MV)	0.06	0.09
$I_+ = (P_B/V_+)/[c_R e^{-\pi 0^{\sigma} 01} f_{10} e^{-\pi + \sigma_{10}} F_1]$ (A)	560	373
$P_+ = I_+ V_+$ (MW)	33.6	33.6
Pulse length	dc	dc
Number of beam injectors	4	4
Number of ion sources per injector	2	2
Ion beam per source, $I_+/8$ (A)	69	46
Plasma ion emission current density, J_0 (A/cm ²)	0.300	0.245
Type of grids	Multiperture	Multiperture
Grid transparency, τ_g	0.45	0.45
Ion beam current density, $J_+ = \tau_g J_0$ (A/cm ²)	0.135	0.110
Grid area per source, A_g (cm ²)	519	424
Initial beam diameter, $2 r_{0 \max}$ (m)	0.257	0.232
Maximum initial beamlet 1/2 - angle divergence, $R_{0 \max}'$ (rad)	0.030	0.030

TABLE V-6. Neutral Beam Injector Design Summary (Continued)

	D^O	T^O
Transverse emittance, ϵ (n-rad)	3.9×10^{-3}	3.45×10^{-3}
Assumed k value (see Fig. V-2)	1.1	1.1
Assumed direct converter efficiency, η_1	0.7-0.8	0.7-0.8
Assumed direct converter efficiency, η_2	0.7-0.8	0.7-0.8
Overall neutral injector efficiency, ϵ_{NI}	0.56-0.59	0.56-0.59
Gas efficiency, η_g	0.45	0.45
Gas input for 8 sources, Q_{in} (Torr-liter/sec)	156	105
Gas input for 8 neutralizers, Q_N (Torr-liter/sec)	56	31.7
Gas equivalent of neutral beam from 8 beam lines, Q_O (Torr-liter/sec)	25	16.7
Tritium feed rate to 8 sources, $0.75 \dot{m}_g$ (gm/hr)	-	$0.75 \times 122 = 91.5^a$
Tritium feed rate to 8 neutralizers, $0.75 \dot{m}_N$ (gm/hr)	-	$0.75 \times 37.8 = 28.4^a$
Tritium to torus for 167 A of T^O , $0.75 \dot{m}_T$ (gm/hr)	-	$0.75 \times 18.9 = 14.2^a$

a. A duty cycle of 0.75 is assumed.

a set of flanges designed for a remote disconnect. The flanged connection is followed by a rectangular flexible bellows and a gate-type isolation valve with a remote operator. The dual beam injector follows at this point and at present is in an early stage of conceptual design.

F. Neutral Beam Test Facility

This test facility can be seen in Figs. XI-1 and XI-2. A shielded cell fitted with an independent power supply, diagnostic equipment and beam dump is provided to test the neutral beam injectors as required. Floor rails and the overhead crane are used to maneuver the injectors in and out of the testing area. While in the test area, both floor robots and an overhead crane mounted robot will serve to hook up, repair and maintain the unit.

Power supply equipment will be maintained dominantly outside the containment building. The very high voltage equipment and final hook-up connections will be permanently fixed in an isolation area adjacent to each of the neutral beam injectors. Lead line lengths from the injectors through the containment shell to the main power supplies have been minimized. Space allocations for cryogenic equipment for neutral beam vacuum cryogenic equipment for neutral beam vacuum cryosorption panels have been made. Rotating machinery for the vacuum system will be placed in the attached power equipment building whenever practical.

G. Neutral Beam Vacuum System

Pumping requirements for each neutral beam injector are divided between two chambers, Chamber I and Chamber II. Chamber I is on the inlet side of the neutralizer and Chamber II is on the exit side of the neutralizer.

The D_2 gas load in Chamber I is calculated to be 45.4 Torr-liter/sec. Assuming an average pressure between 2 and 3×10^{-5} Torr as adequate in Chamber I, the required pumping speed must be about 2×10^6 liter/sec. This pumping speed requires 40 m^2 of cryosorption panel. In Chamber II, the gas load will be 12 Torr-liter/sec. With an average pressure of 1×10^{-5} Torr, the pumping speed required will be 1.2×10^6 liter-sec. Twenty-four (24) m^2 of cryosorption panel will be required to provide this speed. A total of 64 m^2 of cryosorption panel will be used for each beam line. Pumping requirements for T^0 injectors are somewhat less (see Table II-8).

Impedance of the gas flow to the panels must be avoided, and this is accomplished by placing the panels within the neutral beam enclosure itself. Utilizing the vertical side walls of this enclosure appears most appropriate. Dividing the pumping surface equally between both sides of the enclosure will ensure more uniform pumping.

Regeneration of the cryosorption panels will be necessary periodically. To avoid interruption of reactor operation at this time, cryosorption panel area can be doubled and the means provided for isolating half of the area for regeneration while the other half is pumping. One way of accomplishing this would be to put half the pumping panels on movable vertical walls and the other half on the neutral beam enclosure vertical side walls. When the movable walls are down, they isolate the enclosure side walls, thus permitting them to be regenerated. When the movable walls require regeneration, they are pulled upward into pockets where they are regenerated. Pulling them upward exposes the enclosure side walls, which have been regenerated and are ready for pumping.

H. The TETF as a Neutral Beam Injector Technology Demonstration Facility

The experience gained with TETF neutral beam injectors will provide valuable and needed design input for power-producing reactors such as EPR. This technology can be demonstrated in several areas: the separation of beam components, the optimization of beam lines to maximize transport efficiency, thus minimizing ion beam and gas requirements, the technology of direct energy recovery, and the matters of lifetime and reliability of various source, accelerator and injector materials and components in dc reactor operation.

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VI. TETF Toroidal Vacuum System

Optimal performance of the plasma in the CIT mode placed quite stringent demands upon the toroidal vacuum system. A "conventional" vacuum system has been examined and determined to be unable to meet these demands. The conventional system is presented here to quantify the problem. Several alternatives are discussed.

The conventional reactor vacuum system pumping arrangement is shown in Figs. VI-1 and II-1 and is summarized in Table VI-1. Twenty-four ports of 0.75 m diameter positioned vertically between magnets, both above and below the vacuum vessel, provide access for pumping. Each of the ports is fitted with a plug valve, extension Y-type duct, and a set of two cryosorption pumps--each with an isolation valve. Only one pump at a time is in service, while the other is being regenerated. The system is supported from either the magnet support/torque frame or the floor base. All associated components are designed for remote disassembly and maintenance.

The vacuum vessel nozzle is joined to a shield-plug support housing which includes a flexible (bellows) alignment joint. All materials exposed to the plasma chamber are stainless steel. The duct is placed into position after the shielding blocks are in position, and seal-welded to the vacuum vessel. A shield plug ~ 0.85 m diameter by 0.9 m long, complete with cooling, appropriate shielding materials, and outer piston-type valve plug are mounted in the center of the housing, forming an annular path for gas extraction. The shield plug assembly has as its outer fitting a housing containing the plug valve seat and a remote flange connector to the cryosorption pumps. All parts are designed for ease of maintenance and replacement. Ancillary systems include the valve actuator power supply, coolant lead lines and seals, annular shield plug support, and supplementary shielding.

The shield valve component is followed by a Y-extension duct, an intermediary unit used to maneuver between the surrounding toroidal coils to the dual cryosorption pumps. This unit is flanged at all three ends and supported with positioning brackets.

The cryosorption pumps are parallel mounted on remotely adjustable supports fastened to the magnet support/torque frame (upper units) or floor supports (lower units). Each pump has an isolation valve, flexible alignment membrane (bellows) and a mating remote coupling flange. Ancillary equipment includes the cryogenic supply leads, supplementary roughing pumps and blower connections, regeneration heaters, and system lines for containment and recovery of fuel.

Figure VI-1. TETF top plan view

All major vacuum equipment will be located within the reactor containment shell, with the exception of the helium compressors. This is a safety measure to minimize hazards.

TABLE VI-1. TETF Toroidal Vacuum System Summary

Volume	58.5 m ³
Surface area	131 m ²
Number of pumping ports	24
Diameter of port	0.75 m
Number of cryosorption pumps	24 (48 for on-line regeneration)
Pump speed (for hydrogen) - pump	3500 l/sec (rated)
Effective speed (duct conductance for tritium)	2675 l/sec
Pumpdown time from 10 ⁻³ to 10 ⁻⁵ after burn cycle	10 - 15 sec
Secondary Pumps:	
Number of 1,300 CFM blower stations	6
Number of 1,400 l/sec turbopumps	6
Continuous Pumping During Burn:	
Gas load	60 Torr l/sec

Optimal performance of the plasma requires a neutral density in the scrape-off channel that will not exceed 3×10^{11} atoms per cubic centimeter (Chapter 1.1-B). This density corresponds to a pressure of 1×10^{-5} Torr at room temperature. Such a pressure with a gas load of 60 Torr l/sec requires a pumping speed of 6×10^6 l/sec--a speed much greater than can be expected by the above system.

Conductances and pump speeds depend linearly on the mean molecular speed, and thus scale as $\sqrt{T/293}$. If we consider the average energy of the molecules as 10 eV ($\approx 116,000^\circ\text{K}$) and the existing entry into the duct (with valve removed) as an aperture with a diameter of 0.7 m, the conductance is:

$$C = 3.64 A \sqrt{\frac{T}{M}} y$$

A = Area in cm²

$$A = \frac{\pi d^2}{4} = \frac{(\pi) (70)^2}{4} = 3848 \text{ cm}^2$$

T = 116000°K

$$C = (3.64) (3848) (170) (0.86)$$

M = 4

$$C = 2,051,322 \text{ l/sec}$$

y = 0.86

With appropriate pumping surfaces, it seems possible to maintain the required density--providing the means is made available above the aperture to de-energize fast neutrals so that they can then be pumped. The pumping speed required above each aperture to collect the neutrals would be $\frac{6,000,000}{24} = 250,000 \text{ l/sec}$. Five m² of cryosorption panel could provide this speed. Interposing another set of water-cooled baffles, however, to handle sputtered particles would impede the flow to the panel. Assuming a transmission probability of 0.25 for the water-cooled baffles, 20 m² of cryosorption panel would be necessary above each aperture.

Rough calculations suggest that the power required to maintain a 4°K temperature on the cryosorption panels in the face of high neutron flux would be prohibitive. It may, however, be possible to use gettering surfaces such as titanium or zirconium to accomplish the pumping. The energy entering the gettering surfaces could be used to maintain the surface temperature at the level required to activate the getters for efficient capture. Water-cooling of the walls or interposing baffles in front of the walls may be necessary to control or maintain their temperatures at the most efficient level. A separate source of heat would also have to be designed into the gettering surfaces to initially activate them and also to regenerate them.

The experience gained from operating the TETF vacuum systems would be useful for the EPR and other advanced fusion reactors, primarily in demonstrating techniques required to pump high gas loads. It is expected that the pumping speed required for TETF is an order of magnitude greater than for EPR; but if the vacuum vessel internal design, first wall materials and/or neutral beams contribute higher than expected gas loads, the TETF experience would be invaluable.

VII. TRITIUM PROCESSING AND CONTAINMENT FOR THE TETF

With the exception of a set of unique tritium handling problems associated with the T^0 neutral injection system, the tritium processing and containment requirements and objectives for the TETF are largely the same as those previously described for a Tokamak Experimental Power Reactor (TEPR) in Chapter VII of reference 1. If blanket processing technology is developed, limited testing of promising blanket concepts with tritium breeding and recovery could be accomplished in TETF. However, blanket tritium recovery is not considered to be a mainline design feature of the TETF facility and will not be discussed further in this preliminary scoping document. Instead, the remainder of this chapter will focus on descriptions of design requirements for the major tritium handling subsystems that are essential to sustained TETF operation under inherently safe and reliable conditions.

A. Tritium Inventory and Reactor Fueling (Logistics)

Except for a small quantity of tritium that may have to be admitted to the torus during the portion of the startup phase where a low density target plasma is established, all of the tritium entering the plasma during the burn cycle is supplied by the eight T^0 neutral beam lines. The injection requirements of 170 A of T^0 at 90 keV coupled with a 75% duty cycle leads to a figure of 15 gram/hour for the total tritium delivered to the torus during periods of continuous operation. As was pointed out in a previous section (Chapter V) the tritium input to the injector system that is needed to sustain a 15 gram/hour T^0 injection rate is on the order of 120 gram/hour (when T_2 is used in the neutralizer as well as the injectors). If this 120 gram/hour is evenly distributed over the eight beam lines and the beam line pumping system is turned over once an hour, the average amount of tritium tied up in any one beam line could be ~ 15 grams.

In order to develop a figure for the total tritium inventory needed to support sustained operation of the TETF, it is necessary to make several assumptions about operating practices. The tritium collected by the neutral injector pumping systems will probably not be contaminated significantly by one, two, or perhaps even a dozen cycles through the injector; therefore,

processing of the tritium recycled from the injector pumps (to remove H, D, and other impurities) may have to be performed only once or twice a day. Taking the case of twice a day processing and assuming that the injector system cleanup can be accomplished in three hours, one processing unit would contain 15 grams of tritium at any time and would service four injectors as shown in Fig. VII-1. Allowing for a loaded stand-by processor, a tritium reserve in storage equal to the injector system capacity, and ~ 360 grams of tritium in the plasma exhaust processing system (~ 24 operating hours worth) we arrive at the approximate tritium inventory disposition shown in Table VII-1.

Table VII-1. Tritium Inventory Disposition for the TETF^a

Location	Tritium Inventory (grams)
Neutral Injector Systems (8)	~ 120
Processors for Injector Recycle (3)	~ 45
Plasma Exhaust Vacuum and Processing System (2)	~ 360
Contingency Reserve Storage (1)	~ 120
Total Inventory	~ 650

^aNumbers in parenthesis indicate the number of such systems.

Because tritium breeding is not a design feature of the TETF, the initial tritium inventory and subsequent tritium shipments needed to offset the burn-up will have to be supplied by an outside vendor. Superimposing a 67% plant factor on the 75% duty factor, using the projected tritium costs from reference 1 (\$.30 to \$.90 per Curie), and taking a burn-up fraction of 1% per pass, the cost of the initial inventory (~ 650 grams) is between 2 and 6 million dollars, the tritium consumption rate is 880 grams/year, and the tritium make-up cost is between 3 and 8 million dollars per year.

SYMBOLS

= T° Injector

= Injector Vacuum Pump

= Pump Recycler

= Injector Pump
Exhaust Processor

Figure VII-1. Preliminary schematic of the T° injector pumping, recycling, and processing scenario for the TEF.

B. Fuel Processing Systems

Three types of fuel processing systems are required for the TETF: one type each for the T^0 injector pump exhaust, the D^0 injector pump exhaust, and the plasma chamber exhaust. The T^0 injector pump exhaust processing is seen as a two step operation in which the pump effluent (mostly T, with some H, D, and other nonmetallic elements like O, N, and C) is first gettered to remove the nonhydrogenous impurities and then sent to a single cryogenic distillation column where most of the H and D (presumably present as HT and DT) are removed from the top of the column and sent on to an appropriate feed plate in the distillation cascade that is part of the plasma exhaust processing system (PEPS). Most of the tritium (> 97%) that enters the processor comes off the bottom of the distillation column and is returned to the T^0 injector system.

The D^0 injector pump exhaust processor is a three step operation in which 1) the nonmetallic impurities are gettered out; 2) the hydrogen stream (mostly D_2 with some HD and DT) is sent to the first of two cryogenic distillation columns wherein the HD, which is separated at the top, is sent to the PEPS and the bottoms (containing D_2 and DT) are sent to the second column; and 3) the DT, which is taken off the bottom of the second column, is sent to an appropriate feed plate in the PEPS and the D_2 from the top of the second column is returned to the D^0 injector systems. Schematics of the D^0 and T^0 injector exhaust processing systems are shown in Fig. VII-2.

The plasma exhaust processing system (PEPS) contains an impurity removal step wherein nonmetallic elements and other impurities are separated from the plasma exhaust, leaving a hydrogen stream containing roughly equivalent amounts of D and T with some H impurity. Because the TETF is a totally beam driven device, the hydrogen isotope separations requirements imposed on the plasma exhaust are significantly more complicated than those described for the TEPR in a recent report². All of the deuterium and tritium exhausted from the torus must be separated to the extent that 1) the deuterium-rich stream is contaminated with less than 3% tritium, 2) the tritium-rich stream contains less than 3% deuterium, and 3) most of the protium has been removed from both streams. To do this separation will probably require an extensive cascade of columns with several stages of isotopomeric equilibration. A plausible enrichment cascade scenario for the PEPS is shown in Fig. VII-3. Columns 1,

*Solid Lines of Flow Indicated Paths of Major Hydrogen Isotope Throughput.

Figure VII-2. T° and D° injector pump exhaust processing schematic for the TETP (preliminary).

Figure VII-3. Preliminary cryogenic enrichment scenario for the TETF.

2⁺, 3⁺, and 4⁺ together with their equilibration stages provide the 97 atomic % T stream that returns to the T⁰ injectors. Columns 1, 2, 3, 4, 5, and 7 together with their equilibration stages provide the 97 atomic % D stream that returns to the D⁰ injectors. Columns 5, 6, 7, and 8, with their equilibration stages further couple together to minimize the amount of tritium leaving the top of Column 8.

C. Fuel Storage System

The TETF will be equipped with a fail-safe fuel storage system to hold the contingency reserves for the T⁰ injectors. Although this storage system would contain less than 200 grams during normal operation, it will probably have to be designed to hold the entire TETF tritium inventory in order to facilitate extensive maintenance operations on the neutral beam injectors and provide a storage receptacle for the initial tritium inventory just prior to startup. The concrete-barricaded vault approach described in Chapter VII of reference 1 for the TEPR would probably suffice for the TETF as well. This vault would contain several independently controlled getter-type storage cells.

D. Fuel Circulation System

The network of systems that comprise the TETF tritium handling facility are interconnected by a series of ducts, valves, compressors, pumps, and monitors. These ancillary components are herein lumped together under the heading fuel circulation system. Also included in this system is the overall facility-control center which checks and balances the integrated operation of the entire facility. Considerations pertinent to the TETF fuel-circulation system are much the same as those presented in Chapter VII of reference 1 for the analogous TEPR system. For the TETF, this system must service three separate fuel cycle processing systems (for the D⁰ injectors, the T⁰ injectors, and the plasma exhaust) and must couple with the purge jacket and atmospheric clean-up systems.

E. In-Plant Containment System

As was done for the TEPR,¹ consideration is being given to three levels of tritium containment. Where it is possible to do so, construction of all components that come into direct contact with tritium should be made with low-permeability materials (e.g., selected alloys, metal composites, ceramic barriers), particularly in elevated temperature (> 200°C) regions. It may be necessary

to provide an external, closely-fitted jacket with provision for in-jacket purging around those tritium facility components that: 1) are susceptible to failure, 2) operate at elevated temperature, and/or 3) contain relatively large quantities of tritium. In addition, the main reactor hall and the other buildings that house tritium-handling systems will be designed to include hermetic sealing (from the environment) and an inert continuously processed atmosphere.

F. Purge-Processing System

The purpose of this system will be to remove tritium from both the jacketed and whole-room purge streams. Besides removing tritium from these purge streams under normal operating conditions, it would have to be capable of handling all plausible forms of high-level tritium release. The reactor hall atmospheric clean-up system could be housed in an adjacent building, and provision could be made to process the entire atmosphere in this ancillary building as well. Both the reactor building (including the reactor hall and all interlocking service areas) and the auxiliary build housing the main atmospheric clean-up system would be pressure vessels designed to permit evacuation to several psi, so that massive clean-up operations can be performed by repeatedly flushing the respective rooms with air, drawing off the flushed air, and storing it (as compressed gas) for subsequent processing to remove tritium. In this way, the tritium level in contaminated room atmospheres can be reduced to acceptable levels following an accidental release, without having to equip the TETF with a massive, high-instantaneous-throughput, tritium stripping facility.

G. Safety Implications and Environmental Impact Related to Tritium Handling in the TETF

Based upon the preliminary and relatively incomplete design point described in this scoping document, it is not possible to provide a comprehensive analysis of the tritium-related radiologic hazards that would be associated with a device like the TETF. However, an attempt has been made to define the magnitude and impact of several plausible problems that could develop under the design conditions discussed in this chapter and in Chapter VIII;

i.e. the consequences of 1) tritium permeation into a water cooled first wall, 2) a large accidental release of tritium, and 3) the release during normal operation of 100 Curies per day.

1. Tritium Accumulation in the TETF First Wall Cooling Water Circuit

The preliminary analysis of tritium permeation into the first wall coolant was based on the cooling panel design and related operating parameters described in Chapter VIII. Permeation was assumed to occur through the front surface of the panel only (the surface facing the plasma). Because the back surface is at a significantly lower temperature and may either be differentially pumped to keep tritium pressures lower than in the plasma chamber or completely sealed off from the plasma chamber by welding of the entire panel perimeter, permeation through it was neglected. The front surface of the panel was taken to be 0.5-cm-thick, 316-SS and the area available for permeation was assumed to be 10^6 cm^2 . To be conservative: 1) two temperatures that bracket the upper limit of the average peak temperature reached in the 316-SS panel during the burn cycle (i.e. 200 and 300°C) were used in the calculations, 2) the average value of the tritium partial pressure seen by the panel was taken to be 10^{-3} Torr, and 3) a zero tritium back pressure was assumed to exist in the coolant circuit. Permeation rates were determined using the formula:

$$\phi = (A/X) (\sqrt{P_{T_2}}) \cdot \phi^0 \cdot \exp (-Q_p/RT)$$

where

- ϕ = permeation rate in cc(STP)·mm/cm²·hr·atm^{1/2}
- A = Surface area (10^6 cm^2)
- X = Panel thickness (5 mm)
- P_{T_2} = Tritium Driving Pressure (10^{-3} Torr)
- ϕ^0 = Permeation constant (2,150 cc(STP)·mm/cm²·hr·atm^{1/2})
- Q_p = Activation Energy (15,600 cal/mole)
- T = Kelvin Temperature (573 K in Case A and 473 K in Case B)

(The values of ϕ_0 and Q_p for 316-SS were taken from data in reference 1 for hydrogen. For additional conservatism no correction was made for H/T isotope effects.)

Results of the permeation calculations are summarized in Fig. VII-4. The concentrations of tritium in the first wall cooling water circuit (capacity = 10^4 liters) for the TETF, after a period of about two years, are found to be no greater than those reached in the primary (D_2O) circuit of typical experimental heavy water reactors.³⁻⁵ Even for a circumstance where the actual permeation rate is twice that determined at 300°C (Case A), the inventory of tritium in the cooling water should still be manageable using practices employed in existing heavy water reactors. The drainage, consolidation, and disposal of the entire first wall cooling water inventory (10^4 liters) every two to three years should pose no insurmountable problems, and is probably preferable (from an economic viewpoint) to installing a tritium extraction plant³ for the first wall coolant.

2. Accidental Tritium Releases

In order to make a preliminary evaluation of site-boundary dose commitments due to a large accidental release of tritium from the TETF, the assumptions, criteria, and results developed in a previous publication² dealing with the ANL/TEPR were used. Taking the case presented in reference 2 for a location similar to ANL and assuming an instantaneous release of 4×10^5 Curies of tritium ($\sim 2.5\%$ of the total inventory), all as HTO or T_2O , the dose commitments were ~ 24 rem (whole body) at 500 meters from the release point and ~ 8 rem (whole body) at 1000 meters. From 10 CFR 100, the guideline for total body dose equivalent in an accident situation is 25 rem. Neither case (500 m or 1000 m) exceeds this exposure limit, but clearly the greater source to boundary distance is more desirable.

3. Normal Operating Release of 100 Curies per Day

The case of a 100 Curie/day release was again treated in the same way as described in reference 2 for the ANL/TEPR. Based on χ/Q values developed for two typical locations at ANL, the values for the uncontrolled concentrations at 500 and 1000 meters resulting from the 100 Curie/day

Accumulated Tritium in TETF First Wall Coolant, Curies

Figure VII-4. Accumulation of tritium in the TETF first wall cooling water circuit compared to existing experimental heavy water reactors.

release were $\sim 5 \times 10^{-9}$ Ci/m³ and 2×10^{-9} Ci/m³, respectively. Although the uncontrolled concentrations guide (ERDAM 0524, Reference 6) specifies a limiting value of 2×10^{-7} Ci/m³, preference should again be given to locating the TETF farther away from the site-boundary. While the above values satisfy current ERDA requirements, practices presently in force at ANL would require effluent monitoring if the off-site concentration exceeded 2×10^{-9} Ci/m³.

H. The TETF as a Tritium Technology Demonstration Facility

It is of interest to examine the tritium-related research and development items in terms of their singular importance to the TETF and their longer range applications to the CTR program. There are a number of development problems associated with tritium circulation and containment in the T⁰ injector systems. Unless such injectors are required by EFRs, DPRs and future commercial power plants, which presently is not anticipated, it is likely that the rather substantial technology needed to solve these problems will not see significant further use. It is also worthy of note that the isotope separation requirements for the TETF plasma exhaust recycle system are in line with but more stringent than those anticipated for EFRs and subsequent devices. (Nearly complete isotopic separation of the D-T mixture is required for recirculation of the TETF plasma exhaust.) Although similar isotope separations methods (e.g. cryogenic distillation, thermal diffusion) are likely to be employed in all tritium burning devices, the enrichment cascade designs and operating practices used on TETF will probably differ significantly from those required for subsequent devices.* TETF tritium facility features covering 1) non-metallic impurity and debris removal from plasma exhaust; 2) fuel circulation, monitoring, and control; 3) in-plant containment; 4) purge and atmospheric detritiation; and 5) fuel shipping, storage, and retrieval, all appear to be directly relevant to EFRs and follow-on devices. The prospects for performing meaningful tests of blanket-tritium handling, recovery, and recycle on the TETF appear to be excellent. In short, the TETF described herein would seem to demonstrate many of the tritium-related technologies required for future CTR devices.

* One exception to this generalization is the Intense Neutron Source (INS) currently under design at LASL. The INS also requires continuous enrichment of a D-T mixture to produce tritium of sufficient purity for recycle to a T⁺ ion source.

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6. ERDA Manual Chapter, Standards for Radiation Protection, Energy Research and Development Administration, ERDAM-0524 (1975).

VIII. TETF FIRST WALL AND MAGNET SHIELD

A. Introduction

For the purposes of this report, the region between the plasma and the toroidal-field coils of the TETF is assumed to be occupied by a water-cooled vacuum vessel (the first wall) surrounded with stratified layers of shielding material. Because the production of sensible heat is not a major objective of the TETF, no limitations were imposed on minimum operating temperatures. The principal requirements defined for the TETF first wall and shield are to: 1) provide the requisite vacuum environment for sustained plasma operation; 2) protect the superconducting toroidal-field coils from excessive energy deposition and damage by neutron and gamma radiation; and 3) accommodate access penetrations for neutral-beam injection, evacuation, coolant inlet and removal, diagnostics, maintenance and experimentation. In order to examine the feasibility of meeting these requirements, preliminary designs were developed for both the first wall and the shield. These designs were then analyzed in sufficient detail to evaluate the validity of the TETF concept, identify major problem areas and uncertainties, and provide a point of departure for a more comprehensive conceptual design study in the future.

B. Analysis of the TETF First Wall

In this section, an evaluation is made of the constraints imposed on the TETF first wall design by: 1) materials responses to the anticipated neutron, particle and radiation environment as set forth in Chapter III and in Section C-1 of this Chapter; and 2) structural requirements determined by mechanical loading of the vacuum vessel. In the light of these constraints, a preliminary design description is developed for a TETF first wall system that could: 1) be constructed with existing technology; 2) be maintained for a reasonable lifetime; and 3) provide the requisite environment for sustained TETF plasma operation. Hereafter, the thermal and mechanical responses of this preliminary design to the proposed TETF operating cycle are examined in the context of defining first wall performance.

1. First Wall Materials Considerations

On the basis of available fabrication technology, materials property data and cost, type 316 stainless steel is considered to be the

prime candidate for the TETF first-wall structural material. Materials properties of the stainless steel that pose lifetime limitations in the radiation environment relate to both mechanical integrity, e.g., mechanical strength and ductility, and to dimensional stability, e.g., void swelling and creep. Because of the relatively low neutron wall loading ($\sim 0.2 \text{ MW/m}^2$) and the modest wall temperature ($< 400^\circ\text{C}$) the bulk radiation damage to the first wall of the TETF will not be severe.

Void swelling should not present a problem since the proposed TETF operating temperatures are well below the maximum swelling temperatures. The total swelling for five years operation is estimated to be $< 1\%$. The $\sim 100 \text{ ppm}$ helium generated in the steel after five years is not expected to significantly affect the swelling. The loss of ductility, which is a major concern for stainless steel in a power reactor, does not appear to be critical for the TETF because of the lower wall temperature and limited neutron fluence. A uniform elongation $\geq 5\%$ is retained even after a five-year lifetime. The strength properties of the steel also remain high at the low operating temperature. Thermal creep should be insignificant and a predicted radiation creep of $< 2\%$ appears acceptable. First wall temperatures in the range 200 to 300°C should be adequate to permit hydrogen to diffuse out of the steel thus alleviating any hydrogen embrittlement problem.

A major materials problem that will be particularly critical in the TETF is related to thermal fatigue of the first wall. Because of the characteristics of the plasma, an unusually large fraction of the plasma energy will be deposited in the surface regions of the first wall. The cyclic nature of the plasma burn and the high surface energy deposition will produce relatively large cyclic thermal stresses in the first wall. Careful attention to this problem will be required in the design if acceptable lifetime is to be achieved. Sections B-2 and B-5 of this chapter deal with the first wall stress and strain problems in greater detail.

A low-Z coating on the stainless steel first wall is proposed as a means of controlling the high impurity concentrations in the plasma. This concept has also been proposed for the ANL/TEPR design.¹ Potential low-Z coating materials include beryllium, beryllium oxide, boron, boron carbide, boron nitride, graphite and silicon carbide.

Bulk properties of major concern for the coatings include thermomechanical, chemical and radiation stability. As discussed above, the first wall of TETF will be subjected to rather severe thermal cycling. The coating must maintain its adherence and mechanical integrity under these thermally induced cyclic stresses at elevated temperature. The chemical stability of the coating with both the stainless steel substrate and the hydrogenous plasma should be adequately demonstrated prior to use. The relatively high radiation-induced gas production (helium and hydrogen) rates, which are common to the low-Z materials, may influence the integrity of the coatings and, hence, limit their useful lifetimes. Optimization of gas release mechanisms by selection of appropriate microstructural and morphological characteristics should help to alleviate this problem.

Surface radiation effects on the first wall of the TETF are a major concern because of the high particle fluxes (particularly D and T) to the wall that result from the relatively poor confinement. Physical sputtering of the first wall by deuterium and tritium will be a major source of plasma contamination. Calculated erosion rates of a low-Z (Be) and high-Z (Fe) material produced by physical sputtering from deuterons with monoenergetic and Maxwellian energy distributions are given in Fig. VIII-1 as a function of average energy. The predicted erosion rates for the Maxwellian energy distribution, which more nearly represents the actual case, are decreased slightly from the monoenergetic curves in the energy range of primary interest, viz., 0.1 to 1.0 keV. Since the effects of plasma impurities are strongly dependant on their atomic number, one would conclude that superior plasma performance could be achieved with a low-Z coating. It is anticipated that exfoliation resulting from helium implantation in the surface can be maintained at acceptable levels by appropriate microstructural preparation. The gas desorption problems encountered in present day physics devices will be substantially different and probably less severe for the TETF because of the higher wall temperature.

2. Stress Analysis of the TETF Vacuum Vessel

In order to obtain an estimate of the wall thickness required to withstand mechanical loadings of the TETF vacuum vessel, the toroidal shell with its oval cross section (Fig. VIII-3) was approximated as a straight

Figure VIII-1. Energy dependent physical sputtering yields of iron and beryllium irradiated with D^+ ions that have monoenergetic and Maxwellian energy distributions.

Figure VIII-2. Shell stresses

Figure VIII-3. TETF vacuum vessel (material = stainless steel) 1 of 2

Figure VIII-3. TET vacuum vessel (material = stainless steel) 2 of 2

elliptical shell having a similar cross section. This shell was assumed to have a uniform-thickness wall and to be supported along the top. Fig. VIII-2 shows stresses at the top of the shell as a function of wall thickness for a 2.2 m by 1.7 m elliptical cross section. These include the bending stress resulting from the weight of the shell, the bending stress resulting from external atmospheric pressure, the membrane stress resulting from the same pressure, and the total of the three stresses. Analogous results are also shown for a circular shell with the same cross-sectional area. Two conclusions can be drawn from Fig. VIII-2: 1) stress levels in the elliptical (or oval) cross section are much higher than those in the equivalent circular cross section; and 2) a wall thickness of about 4 cm would be required if the design stress is about two-thirds of the yield stress. Consequently, a much thinner shell wall supported by rings and stringers would be a more practical structural configuration for the TETF.

Structural stability is often a limiting factor in thin-shell design. Buckling pressures for ring-stiffened elliptical shells under external pressure are given in reference 2 as a function of three dimensionless ratios derived from the major and minor diameters of the cross section, the wall thickness and the distance between rings. For a major diameter of 2.2 m, a minor diameter of 1.7 m, a wall thickness of 6.4 mm (1/4 in.), and a 1 m distance between rings, the buckling pressure is 53 psi, which is about three times the design pressure. This is a reasonable factor of safety for analytically determined buckling loads for shell structures.

3. Design Description of the TETF First Wall

The stainless steel toroidal vacuum chamber shown in Fig. VIII-3 is made up of 24 segments whose cross section is an oval, 2.2 m in height and 1.7 m wide. Each segment forms a truncated curved 15° toroidal section bounded at each end by grooved flanges. The sections are clamped together to form an irregular torus whose major diameter is 6 m. The walls of the chamber are formed by two layers of integrated coolant panel as shown in Fig. VIII-3. The flat plate section of the interior panel is ≥ 6 mm thick, faces the plasma, and is covered with the low-Z coating. These inner panels carry none of the structural load and do not provide the primary vacuum boundary. This principal function is to absorb and remove the radiation, particle, and most of the neutron thermal loads that

strike the first wall. The flat plate section of the outer panel is also ≥ 6 mm thick and faces away from the plasma. These outer coolant panels absorb only neutron energy. They are welded to the mating end flanges, are cross-braced with spars, and as such provide the structural support and primary vacuum boundary. Depending on considerations related to thermal stresses and resulting strains, the inner panels can either be bolted to the flanges and spars or completely welded into position. The overall thickness of the first wall assembly, including both inner and outer cooling panels, is ~ 4 cm.

The support of each first wall segment (~ 3500 lbs.) is made through the base of the surrounding shield blocks via the spars. The flanges used to join the segments contain both ring seals and a bellows membrane seal. As shown in Fig. VIII-3, dual seal rings are used to align the segments and to provide a high vacuum seal. A third hard seal is made by the flexible metallic membrane that is welded in position after the flanges on adjacent segments are joined. The flanges are held together by a segmented clamp ring as shown in Fig. VIII-3. Both the seal welding of the bellows and the installation of the clamp rings can be done remotely with existing state-of-the-art technology. Access ports are provided at two positions in each set of flanges so that the seal effectiveness can be monitored and differential pumping can be applied to offset low level leakage. Two of the 24 flanged joints connecting the first wall segments are specially treated with electrically insulative coatings on all contact surfaces to provide a first wall current breaker. Likewise, the bellows membrane in these current-breaker joints is formed by two insulated lapped sections.

All access ports are designed with reinforced nozzles and appropriate cooling systems. Nozzle extensions will be bolted for strength and seal welded on the inside of each penetration to prevent leakage. Major penetrations consist of: 1) twenty-four 0.75 m diameter holes located vertically and equally spaced between the toroidal-field coils to accommodate the reactor vacuum system; 2) eight rectangular ports, 0.55 m wide by 0.8 m high, located on the horizontal midplane for neutral beam injection; and 3) four 0.8 m diameter ports located on the midplane to allow convenient access to the vessel for experimental apparatus or inspection of equipment. In addition, approximately 40 m² vessel wall space is accessible for penetrations housing experimental assemblies.

As described above, both the inner and outer walls of the first wall structure are fabricated with integral coolant panel systems. The coolant panel walls have header channels paralleling the flange ring with supply and return coolant nozzles extending to the outer surface of the vessel. Parallel coolant channels extending from flange to flange remove heat generated in both the inner and outer panel walls of the vessel. The flanges and spars have individually tailored coolant systems and each segment is cooled independently of the others.

Provisions are made for both space allotment and cooling of plasma limiters. The specific design has not been investigated in detail, however, circumferential (poloidal) strips of limiter material containing a high performance cooling system can readily be mounted in the vicinity of the segment flanges. Both access space and coolant are also available for neutral beam dumps. Specific design has not been made but no acute difficulties are foreseen even to the extent of having some on-line adjustment potential.

This preliminary scoping design of the TETF first wall has been predicated on concepts and features that would ease assembly, maintenance and repair operations. The four 0.8 m diameter entrance ports are intended to accommodate mobile inspection and limited repair machinery. The vacuum vessel itself could be disassembled in a straight-forward manner, such that the time interval for removal and replacement of a vessel segment would be less than three weeks. Planned replacement of an experimental vessel segment could be carried out in the same way and on much the same schedule. There is some need for development of in-situ diagnostic instrumentation and remote inspection equipment to aid in the location and examination of structural or mechanical faults. Some of the major assembly and repair items required to support maintenance operations on the TETF are: 1) mobile in-vessel inspection vehicle with remote television and related diagnostics, 2) automatic flange seal welding and cutting machinery, 3) automatic flange clamping and declamping mechanisms, 4) an assortment of special welding and cutting tools for the various types of penetrations, and 5) heavy duty manipulators for the cooling panel sections and other vacuum vessel components.

4. First Wall Thermal Analysis for the TETF

A transient heat transfer analysis of a coolant panel configuration similar to that shown in Fig. VIII-3 has been carried out for a

45 second burn pulse followed by a 15 second exhaust and replenishment phase, as described in Chapter III. The flat-plate portion of the inner cooling panel facing the plasma receives heat loads from radiation, particle transport and neutrons. The outer cooling panel and the shield region beyond it receive only neutron heat loads; hence, the inner cooling panel serves to protect the remainder of the first wall and the shield from the consequences of large accidental energy dumps. The results reported in this section were obtained from one-dimensional calculations carried out using a finite difference, time dependent heat transfer code. (Results of preliminary three-dimensional analyses are very similar to the one-dimensional results presented here.)

In keeping with the concept of using a low-Z liner material to reduce plasma contamination problems, the first surface was assumed to be coated with a ~ 100 μm thick layer of graphite. Based on data from reference 3, a value of ~ 500 $\text{Btu/hr}\cdot\text{ft}^2\cdot^\circ\text{F}$ was developed for the resistance to heat transfer at the coating-substrate boundary. This resistance is likely to be a function of the age of the first wall, i.e. the number of burn cycles, but was assumed to be constant for the present analyses. A heat transfer coefficient of 500 $\text{Btu/hr}\cdot\text{ft}^2\cdot^\circ\text{F}$ between the steel and the coolant was assumed. In several early designs the percentage of the backside of the flat plate surface actually contacted by the coolant was small and, as a result, heat removal problems were exacerbated. The design in Fig. VIII-3 allows 75% to 80% of this rear face to be in intimate contact with the coolant fluid. However, for surface coefficients greater than 300 $\text{Btu/hr}\cdot\text{ft}^2\cdot^\circ\text{F}$, the controlling parameter is the conductivity of the steel rather than the interface heat transfer coefficient.

Figure VIII-4 shows the temperature distribution through a portion of the coating/steel interface region on the inner cooling panel. The lower and upper curves represent the temperature distribution at the end of the first and sixth burn pulse, respectively, after an initial cold start-up. In both cases the temperature drop is approximately 55°C (100°F). This information will ultimately be of value in determining the ability of the coating to withstand the thermal cycling.

Figures VIII-5 and VIII-6 show the temperature distribution through the entire surface plate (with coating) after the first burn pulse,

Figure VIII-4. Temperature distribution through the graphite coating and coating interface region of the interior coolant panel after the first (lower curve) and sixth (upper curve) burn pulse.

Figure VIII-5. Temperature distribution through the entire 1-cm thick surface plate (with graphite coating) after the first burn pulse. The lower curve is for locations on the coolant inlet plane and the upper curve is for locations on the coolant exit plane.

Figure VIII-6. Temperature distribution through the entire 1-cm thick surface plate (with graphite coating) after the sixth burn pulse. The lower curve is for locations on the coolant inlet plane and the upper curve is for locations on the coolant exit plane.

for both the coolant inlet plane and the exit plane. The temperature difference through the plate is about 200°C and the temperature difference across the plate is about 10°C. Fig. VIII-6 shows the same type of data for the end of the sixth burn pulse, i.e., after 345 seconds. The data obtained from these calculations will ultimately be used in the thermal stress and expansion computations needed to support the design of the first wall inner cooling panel.

Figures VIII-7 and VIII-8 show the transient temperature histories for the inlet plane and for the exit plane. The upper curve is for the front surface of the coating facing the plasma; the middle curve is for the stainless steel at the interface between the coating and the steel substrate; and the lower curve is for the interface between the steel and the coolant. The most striking point brought out by these histories is that steady-state thermal conditions are reached essentially after the first burn pulse. The temperature variation during any given pulse is approximately 140°C (250°F) with the maximum temperature reached by the coating being approximately 400°C (750°F). The stainless steel reaches a maximum of only 290°C (550°F). The coolant enters the inner panels at 38°C (100°F) and exits at < 100°C (< 212°F).

5. Strain in the Inner Coolant Panels

Coolant panel strains were estimated for the steady-state burn cycle shown in Fig. VIII-7. The upper set of curves in Fig. VIII-9 shows temperature distributions at the coating side and coolant side of the panel for a typical 60 second cycle. Shifting the temperature scale does not alter the strain variations so the strain results pertain to panels at other temperatures as well, if the shapes of the temperature variations are the same. The middle set of curves in Fig. VIII-9 shows the corresponding mechanical strain histories if the coolant panels are restrained against bending and expansion by restraints that are maintained at the average temperature of the panels. The maximum cyclic strain range is about 0.35%. The lower set of curves in Fig. VIII-9 shows mechanical strain histories for panels that are restrained against bending but are free to expand. The maximum strain range here is about 0.15%. The magnitude of the strain cycling can be reduced by reducing the constraints on the panels or by reducing the variation of temperature gradient through the panel with time.

Figure VIII-7. Transient temperature histories at the surface of the graphite coating (upper curve), stainless/coating interface (middle curve), and stainless/water interface (lower curve) for locations on the coolant inlet plane.

Figure VIII-8. Transient temperature histories at the surface of the graphite coating (upper curve), stainless/coating interface (middle curve), and stainless/water interface (lower curve) for locations on the coolant exit plane.

RESTRAINED AGAINST
BENDING AND EXPANSION.
RESTRAINT AT 375°F.

RESTRAINED AGAINST
BENDING, BUT NOT
AGAINST EXPANSION

Figure VIII-9. Interior coolant panel strains for typical burn cycle

C. Analysis of the TETF Magnet Shielding Systems

This section contains the results of a preliminary scoping analysis of the magnet shielding system that would lie between the plasma and the toroidal-field coils of the TETF. Because the design of this system is governed largely by considerations of nuclear performance and materials constraints, these aspects of the TETF shield are discussed first. Following those discussions, an attempt is made to evaluate the thermal behavior of a relatively low temperature shield using energy deposition profiles determined from the nuclear analyses. Finally, guided by the implications and conclusions drawn from the nuclear, thermal and materials studies, a preliminary design description is developed for the shield, which would lend itself to both assembly and maintenance in a practical sense.

1. Nuclear Design of the TETF

A schematic vertical section view of the TETF showing the layout of the first wall and shield between the plasma and the toroidal-field coils is given in Fig. VIII-10. The oval-shaped plasma has a minor radius of 0.55 m and an elongation factor of ~ 1.5 . The noncircular first wall surrounding the plasma allows for an 0.2 m thick scrape-off region and is extended on the outside to provide a test region of ~ 0.2 m for experiments requiring exposure to the plasma.

The space between the first wall and the D-shaped toroidal-field (TF) coils is occupied by the primary (magnet) shield. Since no thermal-to-electrical energy conversion is contemplated for TETF, the principal function of this shield is to protect the superconducting TF coils from excessive energy deposition and radiation damage by neutrons and secondary gamma rays. However, the power density in regions immediately surrounding the first wall is high enough to require a heat removal system similar in many respects to that of a power-producing device. A requirement on the heat removal system is to reduce the operating temperature at the first wall to $< 400^{\circ}\text{C}$ and in the shield to $< 200^{\circ}\text{C}$. This would help to mitigate the effects of radiation damage on structural materials, which would extend the useful life of the TETF as a test facility and reduce maintenance requirements.

Figure VIII-10. Schematic of TTF vertical cross section.

Tables VIII-1 and VIII-2 show the dimensions and material composition of the magnet shield in the midplane cross section. Table VIII-1 is for the inner shield (the shield between the plasma and the inner straight segment of the D-shaped TF coil) along line O-A-B in Fig. VIII-10. Table VIII-2 describes the outer shield along line O-E-F-G-H. Neutronics and photonics analyses were carried out with the calculational methods and nuclear data described in reference 7. The current of fusion (14 MeV) neutrons impinging on the first wall varies in the poloidal direction. For the nuclear, radiation damage and thermal-hydraulics analyses, an average neutron wall loading of 0.2 MW/m^2 was used. A representative value for the duty cycle of 50% was used in calculating time-accumulated effects.

For purposes of the nuclear calculations, the stainless steel first wall is idealized as 3 zones (zones 3, 4 and 5 in Table VIII-1 and zones 4, 5 and 6 in Table VIII-2) of 1, 1 and 2 cm thickness. The intermediate zone represents the coolant channel region with the coolant occupying $\sim 60\%$ of the volume. The coolant in the first wall and shield is water, which has a good moderating power for neutrons. However, in calculating neutron and gamma-ray attenuation, the water was substituted with vacuum to ensure adequate protection of the TF coils and other reactor components in the case of a loss of coolant condition.

A summary of the nuclear performance of the first wall is given in Table VIII-3. The heating in the first wall consists of two components. The first component, which comes from energy deposition by ions, neutrals and electromagnetic radiation from the plasma, is deposited almost entirely near the surface. The second component of the heating comes from energy deposition by neutrons and secondary gamma rays and is referred to as the nuclear heating. The rate of surface heating in TETF is $\sim 25 \text{ W/cm}^2$, which is relatively large because of the 30 MW injection power of D^0 and T^0 and the small energy confinement time. The maximum nuclear heating in the first wall is 2.1 W/cm^3 . After five years of operation with an 0.2 MW/m^2 neutron wall loading and 50% duty cycle, the specific activity in the stainless steel first wall will be as large as 20 Curies/cm^3 .

The first wall is surrounded by a 35-cm thick region of stainless steel (zone 6 in Table VIII-1 and zone 7 in Table VIII-2). This region, together with the first wall, converts $\sim 99\%$ of kinetic energies of

**TABLE VIII-1. Dimensions and Material Composition of the TBT Inner Shield
(midplane section A-B-C-D)**

Zone	Outer distance* cm	Thickness cm	Material Composition	Density Factor
1	55	35	Plasma	0.0
2	75	20	Vacuum	0.0
3	76	1	Stainless Steel	1.0
4	77	1	Stainless Steel	0.4
5	79	2	Stainless Steel	1.0
6	114	35	Stainless Steel	0.9
7	119	5	Boron Carbide	0.9
8	129	10	Stainless Steel	0.95
9	139	10	Boron Carbide	0.95
10	149	10	Stainless Steel	0.95
11	159	10	Boron Carbide	0.95
12	165	6	Stainless Steel	0.95
13	175	10	vacuum and cryogenic shield (50% SS + 50% vacuum)	0.5
14	215	40	TF Coil	1.0

* distance of the zone outer boundary from the center of the plasma.

**TABLE VIII-2. Dimensions and Material Composition of the TTF Outer Shield
(midplane section E-F-G-H-I)**

Zone	Outer distance* cm	Thickness cm	Material Composition	Density Factor
1	55	55	Plasma	0.0
2	75	20	Vacuum	0.0
3	95	20	Test region	0.0
4	96	1	Stainless Steel	1.0
5	97	1	Stainless Steel	0.4
6	99	2	Stainless Steel	1.0
7	134	35	Stainless Steel	0.9
8	139	5	Boron Carbide	0.9
9	141	2	Stainless Steel	0.95
10	196	55	Lead Mortar	0.95
11	200	4	Stainless Steel	0.95
12	210	10	vacuum and cryogenic shield (50% SS + 50% vacuum)	0.5
13	250	40	TFC	1.0
14	260	10	vacuum and cryogenic shield (50% SS + 50% vacuum)	0.5

* distance of the zone outer boundary from the center of the plasma

TABLE VIII-3. Summary of the Nuclear Performance of the TETF First Wall and Shield

Average neutron wall loading	0.2 MW/m ²
Assumed plant factor	50%
<u>First Wall</u>	
Average fusion neutron current to wall	8.96×10^{12} n/cm ² /sec
Maximum neutron fluence (0-15 MeV)	2.25×10^{21} n/cm ² /yr
Maximum gamma-ray fluence (0-15 MeV)	4.94×10^{20} photon/cm ² /yr
Surface heating (plasma and beam ions, neutrons and radiation)	25 W/cm ²
Maximum nuclear heating	2.1 W/cm ³
Atomic displacement	1.1 dpa/yr
Helium production	21 appm/yr
Hydrogen production	52 appm/yr
Specific radioactivity (curies/cm ³ after 5 years of operation)	20
<u>Maximum Values in TF Coil</u>	
Neutron fluence	2.1×10^{16} n/cm ² /yr
Gamma-ray fluence	2.9×10^{15} photon/cm ² /yr
Radiation-induced resistivity in Cu stabilizer (1 MW-yr/m ²)	6.6×10^{-9} Ω -cm
Dose to mylar (1 MW-yr/m ²)	1.0×10^8 rad
Nuclear energy deposition per coil	11 watt
Total nuclear energy deposition (12 coils)	132 watt

neutrons and secondary gammas into heat. The spatial distribution of nuclear heating is shown in Fig. VIII-11. The heating rate falls from a maximum of 2.1 W/cm^3 in the first wall to 0.04 W/cm^3 at the outer boundary of this 35 cm stainless steel zone. The spatial distribution of atomic displacements and helium production in the stainless steel first wall and the first 35 cm of the shield are shown in Fig. VIII-12.

Beyond the 35 cm stainless steel zone, additional shielding is provided to further reduce the nuclear heating in the superconducting TF coil to permissible levels and to attenuate the nuclear radiation to the levels allowed by tolerable radiation damage to the components of the magnet. Because of the well-known restrictions on space between the plasma and TF coils in small size tokamaks, it becomes necessary to use very efficient shielding materials in TETF. Also, since minimizing the overall cost of TETF is a major design objective, it is very desirable to substantially reduce the cost of the magnet shield. This objective is achieved in this design by utilizing the same concept of a two-segment shield similar to that used in the ANL/TEPR design.¹ In the inner side of the torus where the magnetic field is high and the space is most critical, a configuration of stainless and boron carbide is used. Boron carbide pressed to $\sim 95\%$ of its theoretical density is expensive ($\sim \$30/\text{kg}$) but the volume of the inner segment is less than 10% of the total volume of the shield. In the outer region the space restrictions are less severe, and a thicker shield of less expensive lead mortar is utilized. Chemtree lead mortar (1-6-82-5 product) was used. The composition and atom densities of this product are shown in Table VIII-4.

The maximum heating and radiation damage rates in the TF coils (in the absence of penetrations) occurs at the intersection of the midplane with the vertical (inner) segment of the D-shaped magnet projected as point C in Fig. VIII-10. A summary of the neutronics parameters and radiation damage effects at this point is given in Table VIII-3. At 0.2 MW/m^2 neutron wall loading and 50% operating factor, the neutron fluence is $2.1 \times 10^{16} \text{ n/cm}^2$ per year and the gamma-ray fluence is $2.9 \times 10^{15} \text{ photons/cm}^2$ per year. At these fluences, the maximum change in the critical current density of the NbTi superconductor is less than 5% for ten year operation. For the same conditions and ten year operation, the maximum radiation-induced resistivity in the copper stabilizer is $6.6 \times 10^{-9} \Omega\text{-cm}$. The radiation damage effects drop

Figure VIII-11. Spatial distribution of nuclear heating rate in TETF shield for a neutron wall loading of 0.2 MW/m^2 .

Figure VIII-12. Spatial distribution of atomic displacement and helium production rates in TETF shield for a neutron wall loading of 0.2 MW/m^2 and a duty factor of 50%.

TABLE VIII-4. Composition and Atom Densities of Lead Mortar

Element	Composition w/o	Atom Density (10^{22} atoms/cm ³)
H	2.4	3.6
O	3.3	0.3
B	5.0	0.7
C	15.2	1.9
Pb	73.6	0.5
Others	0.5	-

almost exponentially within the magnet with an attenuation coefficient of $\sim 0.12 \text{ cm}^{-1}$. The TF coil conductors are designed to accommodate the maximum expected radiation damage effects with a safety factor of ~ 5 . The radiation damage effects in both the normal and superconductors can be recovered by periodic annealing at near room temperature.

The nuclear heat load in the twelve coils is 132 watts, which is tolerable from refrigeration power and cost standpoints. The mylar insulation will receive a maximum dose of 1.0×10^7 rad per year from energy deposition by neutrons and gammas. According to present information, mylar can be operated under these conditions for ~ 12 years without serious degradation of its dielectric properties. Fiberglass epoxies can stand an order of magnitude higher radiation damage than mylar.

The above analysis did not account for penetration effects. Two types of penetrations are of major concern in the TETF design. These are the vacuum pumping and the neutral beam injection ports. An estimate deduced from work on the ANL/TEPR indicates that the level of radiation at the TF coil will be increased by a factor of ~ 100 due to the presence of these penetrations. A tentative design approach is to surround the beam injection tube wall (outside the shield) with ~ 30 cm of stainless steel. The penetrations will also significantly increase the energy deposition rates in the outer parts of the shield (lead mortar) immediately surrounding the penetrations. It is estimated that ~ 20 cm of stainless steel must be placed between the beam tube walls and the lead mortar (inside the shield) so that the heating rate in the lead mortar can be kept < 0.1 W/cm³.

2. Shield Materials Considerations

The radiation damage level in the shield adjacent to the first wall is ~ 0.9 dpa/year and drops by a factor of 2 every 6.5 cm as shown in Fig. VIII-12. Operating temperatures within the shield are expected to remain below 200°C, and thermal gradients are expected to be low. At these modest neutron fluences and temperatures, the materials properties will not be seriously degraded during the lifetime of the reactor and most of the effects will be restricted to the first few cm of the shield where the neutron flux is highest.

The austenitic stainless steel comprising the first 35 cm of the shield receives most of the radiation energy. The properties of stainless steel that are important in the design are the same as those discussed for the first wall in section B-1 of this chapter. In general, the less severe operating conditions of the stainless steel in the shield will mean that property changes will be less than in the first wall. Swelling, as well as creep and fatigue, should be negligible. The first few cm of stainless steel will undergo some radiation hardening and loss of ductility, but the ductility should remain well above minimum limits. Gas production rates will still be high in the first few cm of the shield. Hydrogen embrittlement may be a problem if the level of hydrogen is allowed to increase above ~ 50 appm.

The next 5 cm of the inner shield, after the stainless steel, is composed of boron carbide, which is a brittle material at low temperatures with a moderate tensile strength and high compressive strength.⁴ The major effect of radiation on boron carbide is the buildup of helium from (n, α) reactions, which can induce swelling and cracking if present in high concentrations.⁵ Neutron radiation can also substantially reduce the thermal conductivity and electrical conductivity.⁶ The degree to which radiation affects the bulk properties depends to a large extent on the amount of porosity present in the unirradiated material.^{5,6} The first cm of the boron carbide layer will produce ~ 500 appm helium during a five year lifetime (at 0.2 MW/m^2 and a 50% operating factor), but the helium production will drop off rapidly after the first cm. This amount of helium is not expected to induce swelling or cracking if sufficient porosity exists to accommodate the gas. The change of thermal conductivity with radiation will have to be factored into the shield cooling design. The materials past this first layer of boron carbide in both the inner and outer shield will receive relatively small neutron fluences, and it is not expected that the materials properties will be adversely affected.

3. Shield Thermal Analysis

The exact physical dimensions of the individual shield modules remain to be determined from analysis of the remote assembly and flux penetration problems; however, for the purpose of the present analysis, which is to establish the feasibility of the shield cooling concept, the shield has been divided into four concentric rings. The heat generation rates are taken directly from the data of Fig. VIII-11. The first 40 cm of the shield is divided as shown in Table VIII-5. The maximum temperature at any point in the shield material was assigned a value of 125°C and the maximum temperature rise in the coolant channels is assumed to be no more than 40°C . These low temperatures were originally selected to ensure more reliable operation. However, as the results in Table VIII-5 show, maintaining these low temperatures and temperature gradients requires relatively large amounts of cooling water, which tends to complicate the design of the coolant delivery and the distribution systems in the shield. A design iteration is required to identify

TABLE VIII-5. TETF Shield - Preliminary Design Parameters for Water Coolant

	<u>0-10 cm</u>	<u>10-20 cm</u>	<u>20-30 cm</u>	<u>30-40 cm</u>
Average Heating Rate, watt/cm ³	1.3	.45	.17	.06
Maximum Blanket Temp. (°C)	125.0	125.0	125.0	125.0
Water Inlet Temp. (°C)	30.0	30.0	30.0	30.0
Water Outlet Temp. (°C)	70.0	70.0	70.0	70.0
Coolant Channel Diameter (cm)	2.54	2.54	2.54	2.54
Coolant Inlet Pressure (atm)	50.0	50.0	50.0	50.0
Total No Tubes	675.0	342.0	156.1	75.0
Total Tube Length (m)	1457.0	755.0	382.0	184.0
Weight (kg)	18,940	21,307	23,675	26,042
Total Weight (kg)	89,964			

more optimum operating temperatures for the shield design that will lead to minimum construction complexity and cost. For the region of the shield nearest the first wall, the operating temperatures may be higher than 125°C, but a design can probably be developed for a maximum temperature of less than 200°C.

4. Design Description of the TETF Shield

The shield for the TETF consists of slab-type blocks made of two types of composite materials, one type for the inner shield and a second type for the outer shield. In addition to its basic shielding function, this slab system is designed for: 1) magnetic field permeability, 2) economic fabrication, and 3) ease of assembly and disassembly for access to both experimental volumes and for maintenance and repair of the vacuum vessel. A minimum of ten major shield blocks are used to cover the vacuum vessel between pairs of TF coils giving a total of 120 pieces. The heaviest piece will weigh less than 18 tons. All shield blocks are equipped with water coolant systems to remove heat and regulate the shield material temperature. Special shielding is provided around the vacuum ports, the neutral beam ports, the major experiment/vessel access ports, and the yet to be designed experimental assemblies. The shield blocks are supported from beneath the reactor.

Of the ten blocks that cover the vacuum vessel between TF coils, one forms the inner shield; it stands vertically and consists of an interior stainless steel slab, 35 cm thick, followed by alternate layers of boron carbide and stainless steel. The top, bottom and outer shield blocks consist of an inner stainless steel slab, 35 cm thick, followed by a thin layer of boron carbide, another stainless steel layer and a canned layer of lead mortar. This shield block configuration, shown in Fig. VIII-10, approximates the contours of the vacuum vessel and is designed to minimize the quantities of expensive materials and the associated fabrication costs. All shield blocks contain graded coolant systems employing deionized water. Piping connections would be made to each individual block through ~ 1.5 in. diameter pipes.

Three columns between each set of TF coils transmit the load from the shield blocks to a steel grid structure located beneath the reactor. These structures are all removable and are provided with electrical conductance interrupters. As shown in Fig. VIII-13, the inner, outer and top shield blocks are supported by the lower blocks in a post and lintel-type configuration. Bolting will be used to tie the shield blocks together, both for structural support and to accommodate counter torque requirements produced by the magnetic forces. The design will include electrical-conductance interrupters between blocks to accommodate flux penetration requirements.

5. Shield Assembly and Maintenance

Major shield maintenance will be performed by removal of the individual shield blocks for repair. The most probable cause for maintenance is a coolant system failure. Other types of reactor repairs may require disassembly of the shielding, particularly in cases where vessel segment removal becomes necessary. Ease of shield removal and assembly is a requirement for a high reactor availability factor. Special equipment for shield removal consists of several sizes of lifting brackets and small remote tools for making coolant connections and decoupling the shield block structural ties.

The following assembly procedure is proposed and referenced to Fig. VIII-13:

Step 1. The lower shield blocks (numbers 1, 2 and 3) are positioned on the support structures and structurally keyed.

Step 2. The inner shield block (number 4) is positioned, keyed and the coolant lines to all four positioned blocks connected.

Step 3. The vacuum vessel segment(s) are brought into position, installed and leak checked. The vessel coolant headers are installed and connections made to the vessel and checked. Outer shield blocks (numbers 5, 6 and 7) are then positioned and structurally interlocked.

Step 4. The upper shield blocks (numbers 8, 9 and 10) are then installed completing the enclosure of the vacuum vessel. Structural and coolant ties are then made.

Figure VIII-13. TETF Shield Block Arrangement

Step 5. Smaller ancillary shielding is then placed around vacuum ports, neutral beam ducts and experimental apparatus as needed.

For removal of the reactor vessel only steps 5, 4 and part of step 3 need be followed.

D. Synopsis of the First Wall and Magnet Shield Performance

Based on the results presented in this chapter, it appears that the major technical features of the TETF first wall and magnet shield can be accommodated with existing or reasonably extrapolated technology. While a substantial development and testing program will probably be required to verify fabrication, assembly, cooling and maintenance concepts in advance of final TETF design, many of the concepts involved will be applicable to EPRs and follow-on devices. As a result, in addition to being a radiation facility, the TETF would also serve as a major test stand for the first wall and magnet shield technology required by subsequent devices. The technologies that would receive in-reactor verification would include: 1) low-Z coatings performance, 2) first-wall heat removal and temperature control, 3) penetration design, 4) vacuum system integrity and maintenance, 5) mechanical response characteristics of the first wall and shield, 6) remote maintenance and assembly, and 7) tritium containment in the reactor coil regions.

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IX. POWER SUPPLIES

A. Ohmic Heating (OH)/Equilibrium-Field (EF) Coil Power Supply

1. Normal Conducting, Copper Coil Design

The reference design for the OH coil and EF coil is based on normal conducting copper windings with an optional superconducting design. The power supplies for the normal conducting coils will be described first. The level of powers for both coils are within the range of operation for pulsing directly off the power line.

The OH coil is designed to provide 4.2 volt-sec to heat the plasma and 5.0 volt-sec to maintain plasma current during the burn phase. The OH coil is charged to 30,000 A in one direction prior to heating and discharged to zero in two seconds for heating. After crossing zero field, the coil current is increased slowly to maintain the plasma current. A full 5.0 volt-sec requirement during burn requires a coil current of 35,700 A. Resistive power loss in the coil reaches 53.9 MW at 30,000 A. The L/R coil time constant is 3.35 sec.

The power supply envisioned to meet these requirements is a phase-controlled, polyphase SCR rectifier driven directly off the power line. A rough schematic is shown for reference in Fig. IX-1 and is not intended as a final design. The main constituents of the power supply are the polyphase transformer (shown here as three phase but, in fact, would more likely be a six or twelve phase system); the SCR rectifier banks; the SCR reversing switches; and an optional filter. The power supply performs in three operating modes: preheating, coil charging mode; the heating, rapid discharge mode, and the burn phase, slow-charge mode. In the preheating coil, charging mode, the rectifier phase angles are controlled to give the maximum positive charging voltage and minimum charging time for the coil. This period coexists with the end of the evacuation and replenishment phase of the reactor cycle. The peak voltage for the supply is determined not by the charging mode, but by resistive voltage drop at the end of the burn cycle. The voltage for 5.0 volt-sec must be 2150 volts. The charging time of the OH coil is given by:

$$t = - \tau \ln(1 - \frac{IR}{V}) \quad (1)$$

Figure IX-1. Representative power supply for OH coil.

Figure IX-2. Typical OH coil current cycle.

where

τ = the time constant of the magnet

R = resistance of the magnet

I = the final current in the magnet

V = the charging voltage

The magnet takes about 6 sec. to charge to 30,000 A at 2150 volts.

For rapid discharge, the firing angles on the rectifier bank are altered to turn on with negative voltage on transformer terminal. This directs power back into the power line from the magnet. This mode continues as long as current flows in the positive direction through the rectifiers. When the current crosses zero, the reversing SCR's are activated and the phase angles on the main SCR rectifiers are returned to forward bias so that slow charge of the OH coil for the burn cycle can begin. At the end of the burn cycle, the rectifier bank is again phased back to return magnet energy to the line. The complete OH current cycle is shown in Fig. IX-2.

The EF coil is designed to operate at 30,000 A and store 68 MJ of energy. The peak resistive power loss in the coil is 21.5 MW, but because the magnet must be able to reach 30,000 A in 2 sec., the possibility exists for drawing as much as 79.3 MW from the line. However, the time for which EF coil power is increasing roughly corresponds to the time that the OH coil is feeding energy back into the line. The system can be designed to take advantage of the energy balance to reduce the peak power demand during this 2 sec. period.

After the EF coil reaches full current, the coil voltage can be reduced to just satisfy resistive power requirements. Again, the power supply would be a phase-controlled, SCR rectifier type similar to that shown in Fig. IX-1, without the reversing circuits. The peak voltage required by the EF coil power supply is 2650 volts.

A typical arrangement for connection of the power supplies, including the neutral beam power supplies, to the power line grid is indicated in Fig. IX-3. In general, the arrangement is rather standard and poses no particular problems.

Figure IX-3. Representative power grid network for TETF power supplies.

2. Superconducting OH/EF Coil Design

Superconducting OH/EF coils are being considered as an optional design. The operating currents and stored energies remain the same for the two coils.

One power supply option operates off the power grid with the same type of system described in section IX-A-1 and shown in Fig. IX-1. Since no resistive losses occur with superconducting coils, the power taken from the grid to the OH coil during magnet charging is smaller than with a normal conducting coil. However, power returned to the grid during the ohmic heating period is much larger because coil energy is not consumed by resistive heating. As a consequence, the peak capacity of the ohmic heating supply must be larger; roughly, 107 MVA, as opposed to 77 MVA for the normal conducting coil. Peak megavolt amperes of the supply can be considerably reduced by switching on a resistor to assist in OH coil discharge during the ohmic heating period. A 0.16 Ω resistor switched into the circuit would reduce the peak power supply capacity from 107 MVA to 43 MVA. This would undoubtedly amount to a substantial cost savings in the power supply but, of course, some cost is required for the resistor switching network.

The EF coil power supply peak capacity would be somewhat smaller with superconducting coils because of the lack of resistive power losses. The required capacity of the power supply for superconducting EF coils is about 67.5 MVA, as compared to 80 MVA for normal conducting coils.

Development of energy storage and transfer systems (EST) of the type that will ultimately be needed for EPR and future machines might be considered for TETF. With EST and superconducting coil systems, the power demand from the electrical grid for the OH/EF coils is reduced to a trivially low amount. EST systems that might be considered are prototype versions of the CCG, ¹ a form of drum-type homopolar generator, or magnetic energy storage with electronic switching for energy transfer. ^{2,3} Such EST systems are not commercially available and still require research and development effort before they can be reliably utilized. Development and utilization of EST power systems on TETF would provide valuable experience for extrapolating to the larger EST systems needed for an EPR.

B. Toroidal-field (TF) Coil Power Supply

Toroidal-field coil power supply represents no major design difficulty. Its size is entirely dependent on how rapidly the coil is to be charged. A 70 kW supply will charge the coil in twenty-four hours and the 425 kW supply will charge the coil in four hours.

C. TETF Neutral Beam Power Supply

The neutral beam power supply must provide approximately 22.5 MW at 60 kV for the D beams and 22.5 MW at 90 kV for the T beams. The power is taken off the electrical grid as shown in Fig. IX-3. Polyphase SCR rectifiers are the most likely choice for developing these voltages. Spark protection probably can be provided by vacuum switch tubes for these voltage ranges.

A high frequency polyphase SCR rectifier with a saturated time delay transformer (STDT)⁴ could be considered as an alternative to the above system. The STDT provides initial spark suppression and power absorption during a fault, and the high frequency supply can be turned off rapidly enough to prevent damage to the high voltage accelerating column. This technique was suggested by Walter Fraeg of Argonne National Laboratory. It requires development and testing before it can be reliably utilized on the TETF. This is a system that would be very attractive for ultrahigh voltage neutral beam systems of the future, such as EPR, and the TETF could provide an interesting test vehicle for this system.

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X. TEST FACILITY UTILIZATION

A. General Considerations

As discussed in Chapter 1, an important objective of TETF is to serve as a radiation test facility that can provide the necessary information required for future developments of tokamaks. The test facility will serve critical experimental needs in several technical areas such as materials, neutronics, thermal hydraulics, stresses, tritium extraction, safety, and environmental impact.

The purpose of this chapter is to discuss the utilization of TETF as a test facility. The primary concern at this stage of the scoping study is to survey the broad types of tests that can be conducted in TETF. No effort is made here to present actual design details for specific experiments. This will need further in-depth studies.

Information will be obtained during TETF operation from measurements made on

- a. reactor components
- b. blanket and shield test modules
- c. sample irradiation (instrumented rigs and other types of experiments.)

The TETF reactor components such as the first wall, magnet shield, penetrations, and magnets are prototypical of EPR and future tokamak reactors. The radiation fields (neutrons, plasma ions, neutrals, and photons) as shown in Tables X-1 and X-2 are very similar to what one expects in a near-term power-producing tokamak. Therefore, measurements of the performance of TETF reactor components will provide a wealth of critical information required for EPR, DPR, and future reactors.

A large segment of the TETF shield will be designed with sufficient flexibility so as to permit experimentation with various types of blanket test modules. These modules will serve the purpose of gaining experience with and examining the viability of multiple choices of structure, coolant, breeding and moderating materials, thermodynamic cycle, and tritium recovery systems.

The third method of the test facility utilization will be through sample (specimen) irradiations. A large space ~ 20 cm thick at the outer part of the torus (~ 40 m²) inside the vacuum vessel and directly visible to the plasma

TABLE X-1. TETF Neutron Source Characteristics

Neutron intensity	7.3×10^{18} n/sec
Neutron power (14 MeV)	16.8 MW
Neutron wall loading, average	0.13 MW/m ²
Neutron wall loading, maximum	0.16 MW/m ²
Average current to the wall (14 MeV)	5.7×10^{12} n/cm ² ·sec
Maximum current to the wall (14 MeV)	7.1×10^{12} n/cm ² ·sec
Maximum annual neutron fluence at 50% plant factor (0-15 MeV)	1.8×10^{21} n/cm ²
Maximum annual gamma-ray fluence at 50% plant factor (0-15 MeV)	3.95×10^{20} photon/cm ² ·sec
Maximum annual neutron fluence at 50% plant factor (13.5-15 MeV)	1.9×10^{20} n/cm ²

TABLE X-2. Power Flow to Wall

Injected Power = 30 MW
 14 MeV Neutron Production = 16.8 MW
 Total Non-neutron Power = 34.2 MW

Radiation Type	Power Flow (MW)	Energy Range	Current (particles/cm ² /sec)
Unconfined fusion alphas	2.2	3-3.5 MeV	3.0×10^{12}
Charge-exchange neutrals	6	10-2000 eV	$\sim 1 \times 10^{17}$
Electron heat cond. and particle flow	14	10-100 eV	$\sim 2 \times 10^{18}$
Line rad., brems., synch. rad.	2	0.01 eV to 100 eV	---
Ion heat cond. and particle flow	10	10-100 eV	$\sim 1 \times 10^{18}$

is available for this type of testing. Special experimental rigs will be designed to allow simultaneous irradiation of a large number of samples with control over the irradiation conditions (exposure to plasma radiation, sample temperature, static or dynamic loading, etc.).

Sections B, C, and D covers the utilization of TETF to serve test needs in the areas of materials, neutronics, and plasma. Section E is devoted to a discussion of the blanket test modules and Section F gives a brief description of the experimental facilities.

B. Materials - Surface and Bulk Effects

The TETF will provide a valuable materials test facility with extensive capability. Materials testing programs can be oriented in two directions: (1) the first-wall components themselves can provide test verification data for a variety of materials and conditions, and (2) adequate space is provided for insertion of instrumented materials samples and test modules. The predicted particle currents and fluences, nuclear heating, gas production, atomic displacements and the first-wall operating temperatures have been presented in the previous section and in Chapter VIII. The TETF will produce conditions at the first-wall that are comparable to those of near term power reactors. In fact, the deuterium and tritium currents to the first wall will be higher than those anticipated in an EPR as shown in Table X-2. Because of the reactor-like conditions, the TETF will be particularly valuable for assessing surface radiation effects on first-wall materials. The neutron fluxes will not be high enough for accelerated bulk radiation tests to high fluences; however, valuable bulk radiation effects data can be obtained because of the appropriate neutron energy spectrum.

1. Surface Effects

Since the first wall of the TETF will be exposed to near fusion power reactor conditions, the first wall itself can be used for materials test verification experiments. The wall will be simultaneously exposed to high photon energy deposition and high particle fluxes, viz., neutrons, ions, and neutrals. Therefore, synergistic effects and combined erosion

rates caused by sputtering, blistering or exfoliation, vaporization, etc., of the selected first-wall coatings can be determined. Since the wall will be operated at elevated temperature (see Chapter VIII, D), the importance of desorption from first wall surfaces can be assessed under conditions of interest. It is anticipated that wall temperature will have a critical influence on the importance of the desorption phenomenon. Since the twenty-four first-wall modules are individually cooled, the effects of some wall temperature variation could be evaluated by adjusting the coolant flow rate in a few panels. In addition, variations in coating preparation or coating composition on selected panels would permit comparison of gas release mechanisms and other surface effects under reactor-type conditions. Use of the first-wall components for materials testing purposes is further enhanced since the design and assembly procedures have been developed to facilitate repair or replacement of the first-wall modules. In addition, adequate space and access have been provided to insert instrumented materials specimens inside the vacuum vessel. By varying the position, cooling capability, and configuration, surface radiation experiments of the type described above can be conducted on a variety of materials under more diverse conditions.

2. Bulk Effects

Both the first wall and experimental test modules can be utilized to obtain data on bulk properties of materials in a fusion reactor environment. Because of the relatively high energy deposition in the surface region of materials exposed to the plasma, the integrity of monolithic and coated first-wall materials can be assessed under thermal cycling conditions at elevated temperatures during irradiation. The effects of temperature can be investigated by varying the amount of cooling provided. Although stresses in the first-wall panels will probably not be accurately known, they should be in the range of interest for future reactors. Special test modules can be used for accurate stress/strain analysis of candidate materials. Analysis of the bulk radiation effects on coated composites will be particularly valuable because the neutron energy spectrum and the ratio of displacement damage rate to gas production rate will be characteristic of anticipated

power reactor conditions. Since the neutron fluxes are only moderate, accelerated swelling or displacement damage tests at high fluences cannot be achieved. However, comparisons of radiation effects produced in TETF at moderate levels (1-5 dpa and 20-100 ppm He in stainless steel) can be correlated with radiation effects produced by various simulation techniques. These damage and gas production levels will be sufficient to assess the validity of the various simulation techniques and extrapolation models used for determining bulk radiation effects. These neutron fluxes will also be sufficient for evaluating the integrity of selected coated composites under simultaneous radiation and thermal cycling.

C. Neutronics

The TETF is an excellent facility for conducting experiments of prime importance for the neutronics analysis and the nuclear design of fusion reactors. At a neutron wall loading of ~ 0.1 to 0.2 MW/m^2 the neutron and secondary gamma-ray fluxes are high enough (see Table X-1) to permit key measurements through the first wall, shield, magnet components, and most of the building walls (biological shield). These measurements are extremely valuable for validating calculational methods and nuclear data. The media for neutronics testing are the basic reactor components, the blanket and shield test modules, and irradiated samples. By proper selection of configurations and materials in the blanket and shield test modules and samples, important information about the nuclear performance of the principal candidate materials for fusion reactors can be obtained.

The neutronics measurements should provide information about all the important neutronics and photonics parameters. Among these are the neutron and gamma-ray heating, tritium production, neutron multiplication, helium and hydrogen production, transmutation, induced activation, afterheat, and biological dose. The presence of major beam and vacuum penetrations provides unsurpassed capability for examining neutron and photon transmission probabilities and streaming through these penetrations. Variation of local shielding surrounding the penetration and examining the effects on reactor components is viable and valuable for future reactor designs.

D. Plasma Physics

The unique advantages that the TETF offers for plasma physics studies result from the long burn time - 1 min. if the OH transformer must drive all the plasma current, and even longer if a portion of the current can be driven by the streaming ions. The beam injectors, of course, will be capable of steady-state operation, and the beams themselves provide the required fueling. (By way of comparison, the TFTR plasma will have burn times of at most 5 to 10 sec, at reduced field.)

The principal plasma physics investigations that take advantage of long burn periods are the following:

- Dependence of long-time impurity buildup on wall materials, plasma profiles, pumping techniques.
- Attainment of steady-state density and temperature profiles, and ion velocity distributions, with beam fueling and high-capacity pumping systems.
- Ability to maintain a moderately elongated plasma cross section for extended periods.
- Maintenance of plasma equilibrium for long periods during which significant currents are associated with toroidally streaming ions.

E. Blanket/Shield Test Modules

Conceptual design studies on fusion reactors have revealed a wide variety of possible combinations and permutations of choices for a blanket coolant, structural, moderating, and tritium producing materials, tritium recovery systems, and thermodynamic cycle. Determining the best possible combinations of these possibilities requires further comprehensive experimental and analytical work. The TETF as presently envisioned is a unique facility for carrying out this work.

The required experiments can best be carried out with test modules. Each test module is designed and constructed with sufficient engineering details to permit a meaningful mock-up of a particular type of blanket

and/or shield. Each module will have its own coolant, tritium recovery system, and possibly its own first wall. Control over operating conditions (e.g. temperature and level of radiation throughout the test module) will be provided for to meet the diversity of requirements in the experimental program.

F. Description of Experimental Facilities

The experimental facilities, although not yet designed, can be classified into five access type categories, each of which would be custom fitted to the surrounding reactor area.

1. Vacuum Vessel - Experiments and tests which utilize a complete vessel module for testing first-wall coatings, coolant systems, limiters, and materials all exposed directly to the plasma for longer irradiation periods.
2. Large in-Vessel Specimens - Experiments and tests requiring vacuum vessel disassembly for their installation and removal yet utilizing the reactor vessel as a support member. Testing of plasma-exposed materials for long periods are contemplated.
3. Small In-Vessel Specimen Facilities - Tubular nozzles extending vertically into the experimental vessel region from both above and below can be made to accommodate small specimens either directly exposed to the plasma or in a cooled tubular enclosure. These facilities offer the advantages of ease of removal during short reactor shutdown. Experiments such as creep testing, cumulative damage studies, activation analysis (rabbit facilities) are possible.
4. Shield Facilities - Large volumes in high flux regions are available in the outer vertical shield region (up to ~ 20 cubic meters). Here shield blocks may be modified to accommodate any number or type of samples, apparatus, or tests that do not require direct plasma impingement. Such facilities are easy to instrument, remove, replace and operate as they need not interfere with reactor systems.

5. Beam Facilities - Penetrations of the shield located between the TF coils can easily be made for many experiments. Diagnostic equipment, neutron and gamma-ray spectrometers, etc., could utilize this type of facility.

Space for experimental equipment has been made available on the reactor room floor and in the surrounding basement area. A large area in the lower level of the power supply building, which is free from radiation or tritium problems, is designated for experimental equipment and on-line data acquisition and computer systems. In addition, a hot cell is available for observing disassembly and analysis of radioactive samples.

XI. REACTOR FACILITY

The overall reactor facility is shown in Figs. XI-1 and XI-2. It consists of the reactor building and seven major support buildings and structures. The housing for the reactor is in a circular reinforced concrete building 150 ft (45.7 m) in diameter by 125 ft (38.1 m) high. A steel liner provides a continuous vapor barrier for containment of tritium and other radioactive effluents. The reinforced concrete walls five feet (1.52 m) thick satisfy both structural and biological shielding requirements. The building contains an 85 ton polar crane and auxiliaries, a neutral beam test area, the reactor structural pedestals and a full lower level for major reactor equipment. During reactor operation the building will be vacated of personnel.

The major reactor support facilities are located in an attached building adjacent to the main reactor cargo doors. This building houses the repair/maintenance hot cell, the mockup/test equipment and other special facilities for initial winding and cryostat preparation of the TF coils. The building has facilities for handling all reactor components prior to their installation. A third large structure also located adjacent to the reactor building houses the reactor magnet power supplies and compressors for the cryogenic units. Adequate cranes and access to equipment are provided. Power leads are run through seals in the reactor containment to the various reactor components.

The facility also has a tritium facility for storage, preparation and reprocessing of reactor fuel. Its central feature is a 45 ft (13.7 m) diameter by 80 ft (24.4 m) high containment vessel providing secondary containment for accidental gas release. Other facility structures include a facility air handling building, an office building housing the control center, a cooling tower array and a 100 MW power substation.

A. Reactor Containment Building

The reactor building is designed to house and structurally support the TETR reactor and its attached ancillary equipment. In addition, it provides: space and equipment for initial reactor assembly and subsequent remote maintenance and repair, an area and equipment for neutral beam testing,

**Figure XI-1. TETF Facility
Plan View**

Figure XI-2. TETF Facility Plan View and Elevation

biological shielding, secondary atmospheric containment, and experimental equipment space. A circular building, 45.7 m in diameter by 38.1 m in height having an elliptical-shaped roof and flat base constructed of reinforced concrete, was selected. The wall, roof, floor and base of the building are 1.52 m thick reinforced concrete, which provides the structural support as well as the biological shielding required. An 85 ton capacity polar crane services the entire reactor floor. This crane is designed for initial reactor assembly as well as subsequent operations with a capacity to maneuver the heaviest reactor component, two assembled TF coils.

A 1 cm thick steel membrane liner provides a continuous vapor barrier primarily for secondary tritium containment. Two cargo-sized sealed doors and two personnel doors (one air lock) are used for access. The reactor support consists of a central pedestal surrounded by an annular ring abounded by the reactor service floor. Both pedestal and ring structures extend downward 6.1 m to a sub-building foundation pad. A sub-reactor basement area houses: the TF coil cryogenic cold box supply, the TF coil energy dump resistors, neutral beam high voltage power source, vacuum pumps and blowers, first wall and shield primary coolant and coolant storage system, magnet power supply cables, and space for experimental facilities. The building is designed to withstand a vacuum to less than 0.5 atmospheres.

Special features of the reactor building include a neutral beam test facility and a remote maintenance manned crane vehicle. The neutral beam test facility utilizes an area on the reactor floor of ~ 10 m by 18 m (see Fig. XI-1). This area is completely shielded such that testing may be performed concurrent with other reactor activities. The test facility is complete with remote handling equipment, a control center, power supply hookup, beam dump, and access to the facility hot cell. A shielded cab mounted on the overhead circular crane bridge is provided for remote maintenance and inspection purposes. This unit is equipped with a small crane, a single heavy capacity and two standard-duty extended reach manipulator arms. It will carry two persons in its shielded cab to a vantage viewing area anywhere above the reactor. Both cab and manipulators will be programmable for routine operations.

B. Mockup Maintenance Building

This building is approximately 150 ft (45.7 m) long 60 ft (18.3 m) wide and 60 ft (18.3 m) tall. It is covered by a traversing crane whose capacity is 70 tons. Three reactor services are performed in this building: first as an initial assembly area for items such as winding TF coils, cryostat assembly, shield block inspection, torus preassembly, etc. A second permanent function is for remote assembly testing and repair support. The third service utilizes a 30 ft (9.2 m) by 40 ft (12.2 m) hot cell for repair and modification of activated reactor components. The hot cell in addition to serving the reactor needs, can accommodate experimental equipment. Equipment in the hot cell includes a crane mounted manipulator and hoist, a rail system for maneuvering neutral beam injectors, a penetration access directly into the neutral beam test area, a normal contingent of conventional manipulators and viewing windows, roof access to the building crane hoist, and a basement storage area access. A laboratory/office wing is provided at one end of the building for maintenance of equipment. A modest machine shop will be included.

C. Power Supply Building

A large building measuring 80 ft (24.4 m) by 200 ft (61.0 m) by 40 ft (12.2 m) tall houses the magnet system power supplies and the compressors for the cryogenic systems. The building is adjacent to the reactor building and equipment is arranged to minimize the power lead runs to strategic components such as the neutral beam injectors. Two overhead cranes traverse the entire building on individual bridges. A partial basement adjoining the reactor building is also included.

D. Tritium Facility

The tritium facility consists of a steel containment building 45 ft (13.7 m) diameter by 80 ft (24.4 m) high and a two-level attached laboratory area of approximately 2000 sq ft (186 m²). The containment building is used for fuel storage and reprocessing equipment. Two air locks allow personnel entry while maintaining the seal integrity. This facility will also house equipment for handling inadvertent release of tritium in the reactor containment building.

E. Office/Control Building

A 60 ft (18.3 m) by 70 ft (21.3 m) building serves as a central control station, office and laboratory space. The building is four stories high having a 30 ft (9.2 m) by 60 ft (18.3 m) control room two stories tall.

F. Air Handling Facility

As seen in Fig. XI-1, a separate building is nested between the power supply building and the office/control building. This facility will handle the reactor building air circulating system. It has a lower air intake plenum and tall exhaust stack. Internally it has isolation equipment for emergency use.

G. Cooling Towers

Four forced draft cooling towers are used to remove the 100 MW of power supplied to the facility. All reactor systems use heat exchangers between cooling tower and heat source to prevent water contamination. Thus the tower system will house secondary coolant pumps and a complete purification system. One tower serves the reactor with a relatively high Δt spread. The other three units run closer to ambient conditions supplying water to cool power supplies, lead cables and other equipment.

H. Power Substation

A 100 MW substation with 132 kV incoming power is provided.

APPENDIX A
REACTOR ASSEMBLY PROCEDURE

The following assumptions are postulated concerning the equipment, reactor components and general conditions existing on the reactor site prior to its initial reactor assembly.

- The TF coils will be fabricated in the mockup area and checked at the rate of one per month using a dual assembly line. Large diameter OH and EF coils will be fabricated within the reactor containment building.
- A full scale one-quarter segment of the reactor is constructed in the mockup area to develop and record the final remote operation procedures required for the installation of each reactor component. All remote handling tools are capable and tested on site.
- The reactor and mockup/maintenance buildings are operational for the reactor assembly (all cranes operate, reactor pedestal complete, roofs completed, etc.) even though final construction is not complete.
- The assembly time is to be held to a minimum but without undue planned overtime. A basic single shift operation will prevail with inspections coming on off-shift hours.
- After each component is installed, a complete test will be made to verify the integrity of the whole system in order to eliminate any cumulative leaks or faults in the assembled portions.

The assembly sequence is depicted in Figs. A-1 through A-8. It must be emphasized that no component will be assembled for which there is no remote assembly machine or procedure previously prepared and tested. Due to induced radiation of components after even a brief period of operation, most repair or disassembly work will require remote handling systems that will have operated satisfactorily before activation. For the initial assembly, the following stepwise activity is contemplated.

Step 1. With a clean reactor base, the lower-central stainless steel TF coil support spindle is placed into position and tied into the foundation (see Fig. A-1).

Step 2. The lower-inner EF and OH coils are then placed on a temporary support bed beneath the reactor after which the center TF coil support ring is installed (see Fig. A-2).

Step 3. The TF coils are to be installed as shown in Fig. A-3. These coils, with their partial cryostat, weigh ~ 35 tons and there is an option to assemble two coils on the reactor floor, inspect and test and install the pair as a unit. All operations will be preceded by an approved remote assembly procedure.

Step 4. The TF coils would be installed at an anticipated rate of four weeks or less per unit. After a three-to-four TF coil installation is complete and tested, the lower shield supports along with the lower and inner shield blocks may then be installed between the completed coil array, as shown in Fig. A-4. TF coils are added to the structure continuously from each open end.

Step 5. As TF coils are installed, the installation of lower and inner shield blocks continues to match the coil progress. Installation of the vessel segments can be started. This will be accompanied by the installation and hookup of the vessel cooling piping manifolds (see Fig. A-5). After multiple segments of the vessel are positioned, assembled and tested, the outer shield blocks may be set, followed by the upper shield. The lower and upper vacuum ports may then be attached to the vessel through the finished shielding (see Fig. A-6). This assembly sequence will be maintained until all TF coils are in position and tested.

Step 6. With the TF coils in place, the outer OH and EF coil support structure will be assembled. Beginning with the lower outer coils, the assembly process will proceed adding upper coils as the lower ones are installed and adding to the support frame structure. During the same interval, the reactor shield, vacuum vessel segments and vacuum port extension duct installation and testing may proceed to completion (see Fig. A-7).

Step 7. The final installation of the inner OH and EF coils and the support frame will be completed followed by the peripheral reactor attachments such as the neutral beam extension ducts and the vacuum pumps (see Fig. A-8). Torque members will be fitted to the magnet support/torque frame at which point final inspection and tests will be made prior to hook-up of power supply leads, cryogenic lead lines and other ancillary external reactor equipment. The last components installed will be the neutral beam injectors.

Upon completion of the reactor there should exist a full scale one-quarter section mockup of the reactor vessel, shield blocks, neutral beam and vacuum

duct extensions together with a complete inventory of operable remote assembly equipment with a set of approved maintenance and disassembly procedures. This mock-up will be required to program and check-out remote maintenance equipment prior to performance of repair operations on the TETF.

Figure A-1. TETF Assembly Sequence

Figure A-2. TETF Assembly Sequence

Figure A-3. TETP Assembly Sequence

Figure A-4. TETF Assembly Sequence

Figure A-5. TETF Assembly Sequence

Figure A-6. TET² Assembly Sequence

Figure A-7. TETF Assembly Sequence

Figure A-8. TETF Assembly Sequence