TETR - A Tokamak Engineering Test Reactor to Qualify Materials and Blanket Components for Early DT Fusion Power Reactors

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The general features of a beam driven tokamak reactor designed to provide engineering information for the Demonstration Power Reactors of the late 1990's is presented. The TETR is based on the anticipated state of the art plasma physics and engineering technology of the 1980-1985 period and as such, should be able to be built and operated in the 1985-1990 period. The reactor provides an integrated neutron wall loading of 1 MW-yr/m^2 per year of operation to a 20 m^2 test area. Details of the materials test program are given along with an assessment of the difficulty of meeting the required goals in the desired timeframe.

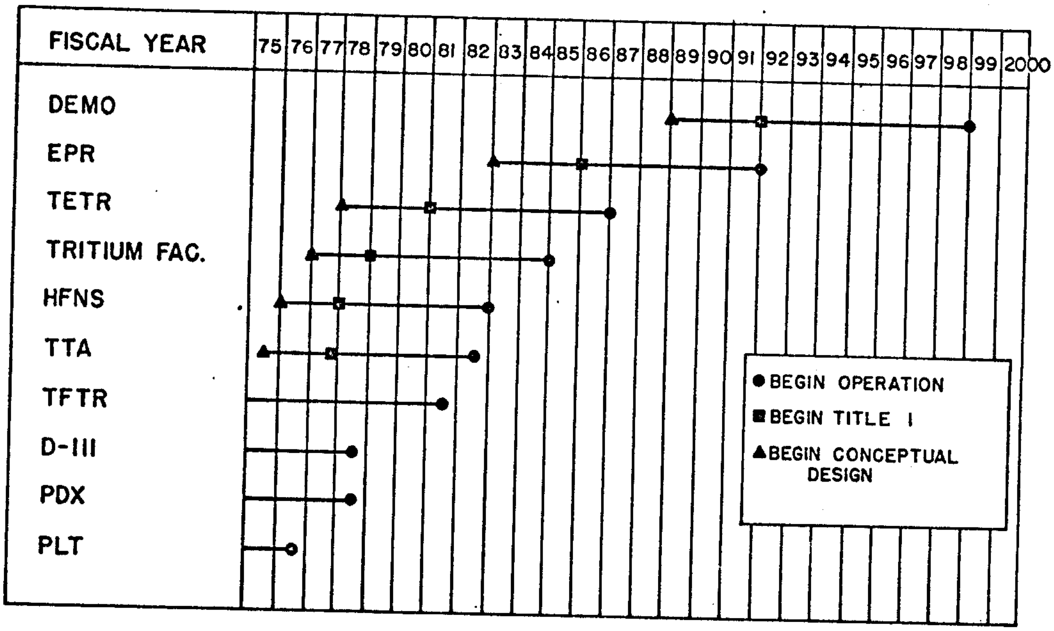
INTRODUCTION

There is an urgent need for a DT fusion device which is mainly devoted to testing engineering components and materials that are being considered for the Tokamak Demonstration Power Reactor (DPR) of the late 1990's. In the present USERDA program plan to commercial fusion power,[1] such testing under the combined influences of irradiation, temperature, vacuum coolant environment as well as the time dependent stresses typical of tokamaks is scheduled to begin in the 1989-1992 time period. Unfortunately, the preliminary design of the DPR must begin in 1989 with Title 1 Construction slated for 1992. This is uncomfortably close to scheduled operation time of the Fusion Engineering Research Facility (FERF) in 1989 and the Experimental Power Reactor (EPR) in 1992. We describe a device in this paper which we believe could be built substantially earlier and with more confidence than the present FERF-EPR concepts because it requires no substantial extrapolation of the state of the art of either physics or engineering beyond the 1980-1995 period. The present paper represents our preliminary conceptual design based on an earlier study[2] and a more detailed report will be issued early in 1977.

DESIGN PHILOSOPHY

The requirement of operation in the mid-1980's means that the Tokamak Engineering Reactor (TETR) must be designed with plasma physics information available in the early 1980's. An example of how we can see the TETR in the overall logic III scheme of ERDA is shown in Figure 1. The major plasma

Figure I TIME FRAME IN WHICH TO CONSIDER TETR (AFTER LOGIC III)



physics input (in addition to the last 20 years of effort) comes from five U.S. devices (PLT, PDX, D-III, ORMAK-UG and TFTR) as well as foreign experiments such as T-10, T-10 M, TFR-II, JT-60, JET, and T-20. One would expect to utilize information from experiments like ASDEX, ISX, Alcator Upgrade, Rector, Microtor, and Macrotor.

The approacn to the design of TETR has been to have at least two solutions to all the difficult technologies such that in the event our base case design does not prove feasible in a 1985-87 timeframe, one would still have a potential solution which could be employed. The major areas where such back-up positions have been investigated are listed in Table 1 and we shall briefly discuss each below.

TABLE 1. TETR Operational Modes

System	Base Case	Alternate Case
Heating	150 keV D^+	200 keV D^-
Plasma Stabilization	Feedback	Inherent
Fueling	Pellets	Beams
Divertor	Unload	Shielding
TF Coil	NbTi	NbTi+Nb_3Sn
Structure	250°C+300°C Anneal	10cm C ISSEC+ 300°C Anneal

Since we cannot be certain that ignition physics will be state of the art (SOA) in 1980-2, and because we would like to build as small a reactor as possible which achieves the engineering goals we have set (for example, high neutron wall loading), the Two Component Mode (TCT)[3,4] of plasma operation has been chosen. The plasma shape has been somewhat elongated to increase beta and the neutron wall loading. Such decisions immediately raise two potential problems, the availability of reasonably efficient and acceptably priced neutral beams, and methods to control vertical displacements of the elongated plasma. Our initial approach to the beam problem is to use 150 keV neutral beams from positive ion sources (TFTR will use 180 keV D^+) with 1/2 and 1/3 energy components separated out. Such beams should be available to PLT and TFTR for ~$0.75 per watt (in 1976 $) and at approximately 50% efficiency. Allowing for some advances beyond 1981, we could use 150 keV D^+ with increased extraction efficiency of D^+ (~90-95%). The greatest advance is to use 200 keV D^- injectors to upgrade the overall efficiency to ~80%.

The feedback stabilization of MHD instabilities on the order of a few milliseconds is already SOA in PLT for horizontal motion and by using the normal VF coils close to the plasma we expect that such control will be available for TETR. Results in Versator[5], Rector[5] TO-1,[6] and PDX should also prove the use of this technology.

The base case for fueling has been chosen to be pellet injection. This concept is currently being tested in ORMAK and will be tested in ORMAK-Upgrade and possibly TFTR. If this concept does not work, we can always fuel with beams and accept the penalty of somewhat lower wall loading, increased capital investment and operating costs.

Impurity control in the plasma is by two methods, one of which may not be necessary if the other is completely successful. We use a double null poloidal divertor in the unload mode to remove fuel and helium escaping from the plasma and ionized impurities sputtered from the chamber walls. The walls are also protected by a carbon curtain liner[8] operating at approximately 1200°C, well above the methane formation temperature. The divertor concepts should be tested in PDX (1978) and carbon liners in ISX (1979).

The TF coils are superconducting of the same size that will be tested in the Tokamak Test Assembly (TTA) about 1982. The use of these coils is not mandatory and if such magnets are not available, it would be possible to use cryogenic Al(4.2°K) or water-cooled Cu at 300°K and pay about $40 M a year of extra power costs.

The TETR blanket and first wall structure will be made of 316 SS and operated at a maximum of 250°C. The object is to operate in a regime where the first wall and blanket structure need not be replaced during the lifetime of the reactor due to radiation damage. Provisions have been made to periodically heat the first wall to 300°C to remove some of the damage accumulated at low temperatures. A back-up position has also been investigated that is to protect the first wall with a 10 cm thick carbon ISSEC.[9] Such an approach would lower the displacement damage by a factor of 3 and the helium generation by a factor of 10. The use of carbon at high temperatures in plasma environments should be tested in PDX or ISX in the late 1970's to early 1980's and high temperature radiation damage should also be available in the same time period from fission reactor irradiations.

GENERAL REACTOR CONFIGURATION

The major features of TETR are given in Table 2. A cross sectional view of TETR is given in Figure 2 and top and isometric views are given in Figures 3 and 4 respectively. The cross sectional view has included the ISSEC to protect the first walls even though in the base case the ISSEC is not required and the first walls are moved out to the position of the ISSEC surface. The overall height of the toroidal field (TF) coils is about 11m and the reactor is approximately 16m in diameter (exclusive of injectors). The normal vertical field (VF) coils are copper and are placed inside the 16 superconducting TF coils while the ohmic heating (OH) coils are operated at cryogenic temperatures. The shape of the blanket closely follows the 2 to 1 plasma elongation and triangular cross section. The total surface area of the chamber walls is 115 m^2 but only 60 m^2 is suitable for testing loops. Sixteen test stations are placed between the 16 TF coils and are located on the upper half of the torus. The bottom half of the torus contains ports for 12 neutral beam injectors and associated vacuum equipment in addition to 12 quick access bayonet test stations. The test modules are approximately 1m by 1m in cross section and 70 cm thick. They can be remotely removed and replaced by simply raising one VF coil and swinging up a movable shield in the torus (Figs. 2,3,4). The final vacuum weld is made at the back of the blanket. The neutron wall loading is ~1.4 MW/m^2 and with an anticipated 70% P.F., approximately 1 $MW\text{-}yr/m^2$ average is incident on the first wall per year of operation.

TABLE 2. Features of the Tokamak Engineering Test Reactor

Operating Regime	Two Component
Fuel Cycle	D-T
Major Radius	3.25 m
Plasma Half Width	0.6 m
Height to Width Ratio	2
Plasma Current	2.5 MA
Impurity Control	Double Null Divertor + Carbon Liner
Total First Wall Area	120 m^2
Max. Accessible Test Area	60 m^2
Active Test Area	20 m^2
Number of Neutral Beams	12
Energy of Beams	150 KeV
Duty Cycle	83%
Magnetic Field on Axis	42 kG
Maximum Magnetic Field	85 kG
TF Superconductor	TiNb
Number of TF Coils	16
Base Structure	316 SS
First Wall Coolant	Water
Maximum Structure Temp.	250°C
Maximum Coolant Temp.	225°C
Thermal Power Output	346 MW
Neutron Wall Loading - Test Stations(P.F.=.7)	1 $MW/m^2/yr.$
Number of Test Stations	16

The injected power to the plasma is 150 MW and, with a Q of 1.78, the average heat handled by the coolant system during a burn cycle is 346 MW. Approximately 157 MW of this heat is absorbed by a water coolant operating at 200-225°C [2.8 MPa (400 psi)] and deposited in cooling towers. Another 39 MW is collected by the helium coolant of the test modules and also rejected to the atmosphere. Finally, ~150 MW is collected in the divertor zones and similarly rejected.

CROSS SECTION VIEW OF TETR

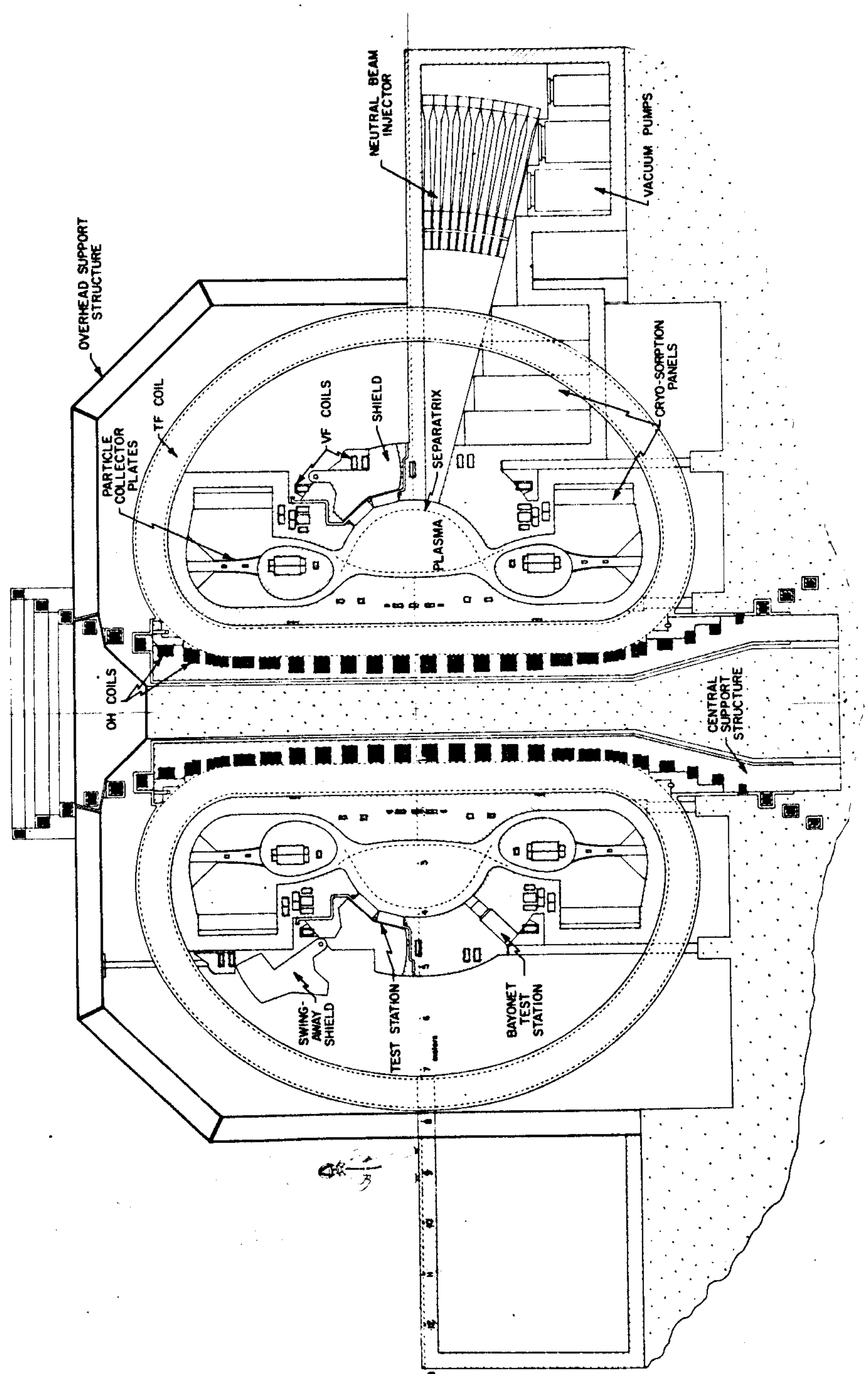


FIGURE 2

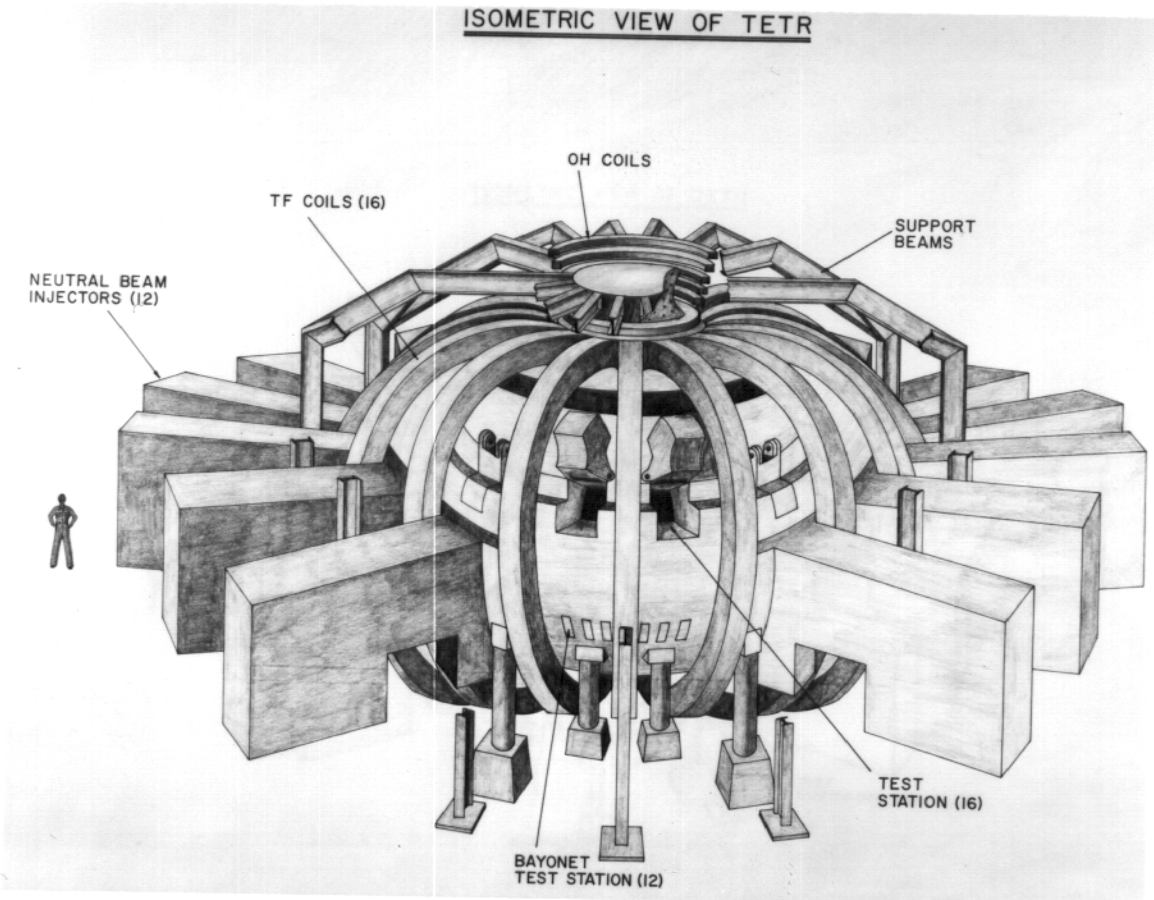


FIGURE 3

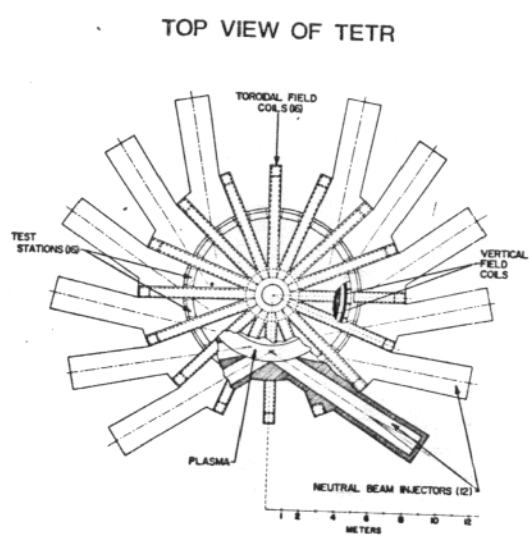


FIGURE 4

PLASMA ANALYSIS

The plasma considerations for a beam driven two component tokamak include plasma start-up, MHD equilibrium, point and space dependent fluid simulation of the burning plasma, plasma control and stability, divertor design and impurity control, fueling, and shutdown. We have analyzed each of these problems but will discuss here primarily the results of the burn cycle analysis, the simulation calculations, and the plasma feedback control system.

The plasma shape shown in Fig. 2 has been determined by MHD equilibrium calculations described in detail in another paper.[10] This particular equilibrium shape has two null points on the separatrix allowing for a double null poloidal divertor and the parameters characterizing this equilibrium are listed in Table 3-a. Several equilibria have been found including shapes that are inherently stable to vertical displacements.[10] However, the shape shown in Fig. 2 has a vertical field decay index of -2 and is, therefore, unstable to vertical displacements. This is not an unusual circumstance and such shapes have been stabilized with moderate feedback system on experiments such as Versator,[5] Rector,[5] and TO-1.[6]

A linear stability analysis of the vertical position has been carried out, including the effects of eddy currents in the conducting wall surrounding the plasma, to determine the necessary characteristics of the feedback system. Without feedback or eddy currents, the force on the plasma in the Z or vertical direction when there is a rigid displacement, ε, in that direction is

$$F_Z = 2\pi R I_p \left.\frac{\partial B_Z}{\partial r}\right|_{r=0} \varepsilon \qquad (1)$$

where r is the radial coordinate, R is the major radius, and I_p is the plasma current. The equation of motion is

$$\frac{d^2\varepsilon}{dt^2} + \omega_o^2\,\varepsilon = 0 \qquad (2)$$

where $\omega_o^2 = I_p B_Z n/(\bar{n} maR)$ where $\bar{n}$ is the density of ions of mass m and n is the decay index,

$$n = -\frac{R}{B_Z}\left.\frac{\partial B_Z}{\partial r}\right|_{r = 0.} \qquad (3)$$

The equation has unstable solutions if the decay index is negative.

Movement of the plasma induces an electric field in the liner and a force which tends to slow down the plasma motion. In a cylindrical approximation, the equation of motion is modified by the eddy currents and becomes

$$\frac{d^2\varepsilon}{dt^2} + \omega_o^2\varepsilon = \frac{\mu_o\delta I_p}{\bar{n}ma} J_o(t). \qquad (4)$$

where δ is the thickness of the conducting wall and $J_o(t)$ satisfies

$$\frac{\mu_o\delta\rho_w}{2}\frac{dJ_o}{dt} + \frac{J_o}{6} = -\frac{\mu_o I}{2\pi\rho_w}\frac{d\varepsilon}{dt} \qquad (5)$$

ρ_w is the radius of the wall. One can obtain a dispersion relation from Eqns. (4) and (5) and assuming a 2 mm stainless steel first wall and TETR parameters, the growth rate predicted for vertical displacements is $1.3 \times 10^3 sec^{-1}$. (These same equations predict a growth rate of $1.2 \times 10^3 sec^{-1}$ at n = -1 for Versator, in good agreement with experiment). This growth rate is quite modest and implies that a rise time of 0.1 msec on the feedback amplifiers should be sufficient. For comparison, the growth rate in the Scyllac theta pinch experiment[11] is $\sim 10^6 sec^{-1}$ requiring an amplifier rise time of less than 0.5 μsec.

The analysis has been extended to include the influence of feedback coils on the dispersion relation. The design value for the gain on the feedback system is 37 G/m, which is twice the minimum value, and provides us with some gain margin. The required feedback field can be produced by making a small change in the vertical field coil currents. ΔI values no larger than 10 KA in eight of the vertical field coils can produce a radial field of 35 G

TABLE 3. a. MHD Equilibrium Plasma Parameters

Major Radius (m)	3.25	Axial Toroidal Field (T)	4.19
Magnetic Axis (m)	3.37	Max. Toroidal Field (T)	8.50
Plasma Half Width (m)	0.60	β_θ	3.80
Height to Width Ratio	2.0	β_{TOTAL}	0.066
Shape Factor	1.53	Plasma Current (MA)	2.52
Aspect Ratio	5.42	q(o)	0.95
Vacuum Vertical Field, B_z°(T)	0.37	q_{max}	2.5
Decay Index, n	-2	Plasma Volume (m^3)	41.6

b. Plasma Parameters from Fluid Simulation Analysis

	Global Model (Base Case)	Space-Time Model
$<n_e>$ (cm^{-3})	7.7×10^{13}	1×10^{14}
$<T_e>$ (keV)	11.3	6.2
$<T_i>$ (keV)	12.2	6.3
Q	1.78	1.36
Γ	0.36	1.0
τ_E (sec)	0.110	0.076
$n\tau_E$ (cm^{-3}-sec)	8.0×10^{12}	7.6×10^{12}
τ_p (sec)	0.092	0.039
Power to Divertor During Burn (MW)	203	190
Fueling Rate (cm^{-3}-sec)	-	2.4
n_T (target plasma)(cm^{-3})	6.3×10^{13}	9×10^{13}
n_D (target plasma)(cm^{-3})	1.4×10^{13}	1×10^{13}
Fractional Burnup %	0.28	

at the magnetic axis. For a 10 cm vertical displacement (very large), the feedback system must provide 773 G which, if delivered in one instability growth time, requires a reactive power of 6.4 GW. Comparing this to the reactive power of 35 MW on the scylla sector experiments,[11,12] we see our reactive power requirements are somewhat larger but the rise time for TETR is much lower. It appears, therefore, that feedback stabilization is feasible and cost estimates show it to be inexpensive.

The plasma parameters during the burn have been estimated using both global and space dependent plasma simulation models using the same scaling laws as in our previous work.[13] The results are summarized in Table 3b. The lower average values of temperature and energy confinement time are due to the shape of the peak density and temperature profiles that result when one models an unload divertor.[14] In addition, one notes from the space dependent simulation in such a divertor mode that τ_E/τ_p ~2. Thus, the particle confinement time is quite short, unlike present experiments with flatter density profiles. Interestingly, a short τ_p permits a high tritium fraction, 90%, even in the presence of intense deuteron beam injection. This is also the most desirable mode for operating a counter-streaming ion tokamak.(15)

The plasma burn cycle for TETR is given in Table 4. The programming of the OH and VF coils during the 1 second current rise to 2.5 MA has been analyzed including both resistive and inductive effects. (A rise time of 50-100 msec, such as is planned for the Tokamak Fusion Test Reactor (TFTR) is also usuable). The maximum power necessary is 563 MW and the flux swing during the startup is 22.2 v-s. During the burn, an additional 1.4 v-s are required.

The burn time for TETR is 60 seconds. From the standpoint of maximum flux swing capacity in the transformer primary, and apart from the plasma physics considerations which might limit the burn time, the length of the burn could be extended several fold. Ultimately, this will be determined by impurity effects, the state of fueling technology, and limits on the peak power requirements.

NEUTRAL BEAM INJECTION SYSTEM

Two systems have been considered for neutral beam injectors. The base case in a 1980's state of the art 150 keV positive ion system with magnetic pre-selection of the full energy component and direct conversion of the unneutralized fraction. An alternative approach would be to use a negative ion source system which would operate with a greater inherent efficieny, even at higher injection energies.

The base case neutral beam system provides 150 MW of injected D° at 150 keV.

TABLE 4. Burn Cycle for TETR

Time (sec)	Phase Description
0-1	Startup: plasma and divertor currents rise to full maximum; transformer currents begin to drop; plasma ohmically heats to T_e = 2.1 keV.
1-61	Burn: gas injection to full density; neutral beam injection to maximum temp.; tranformer currents drop to maximum negative.
61-62.5	Shutdown: beam injection discontinued, plasma cool-down, current in transformer reversed; plasma and divertor currents drop to zero.
62.5-72.5	Recharge: plasma cool-down; residual gas pumped out; chamber refilled with fresh fuel; transformer currents reset to initial values.

The system consists of 12 beam lines arranged as shown in Figures 2 and 3. Each beam line has 20 positive ion sources of the Berkeley type arranged in a 2 x 10 vertical array. The basic parameters of this system are summarized in Table 5. A power flow diagram is shown in Figure 5.

In order to minimize space-charge blow-up, present day neutral beam injectors employ a close coupled gas neutralizer cell following the final extractor grid. Consequently, all species emerging from the ion source enter the neutralizer. The species composition of a Berkeley type source, is typically 75% D^+, 15% D_2^+, and 10% D_3^+. The molecular species (which unfortunately are more efficiently neutralized due to their lower velocities) give rise to half and third energy neutral components which do not penetrate to the plasma interior. In order to increase the overall injector efficiency of the base case system, these unwanted species are rejected at low energy (15 keV) by magnetic selection. The desired atomic component is then post accelerated to 150 keV and neutralized in a decoupled gas cell with separate gas feed.

There are additional advantages to be gained by removal of the molecular species. The power density (a critical design parameter) in the 150 keV accelerator structure can be reduced by 25% with this preselection. The method also results in reduced charge exchange losses in the beam transport line, since these losses depend strongly on the total beam current.(16)

The relatively poor neutralization efficiency of the 150 keV D^+ dictates the recovery of the power in the unneutralized fraction by a direct converter system. (It should be noted that direct conversion is to be used also on TFTR). The direct converter can be of the in line type(17) which does not require a bending magnet following the neutralizer. In the base case design, direct conversion reduced the net power required by the injectors from 533 MW (28% efficiency) to an acceptable 249 MW (60% efficiency).

An alternative approach to the beam system (which would allow higher injection energies at greater efficiency) is based on a negative ion source. Unlike the positive ion system chosen as the base case, this approach would require a substantial development of technology.

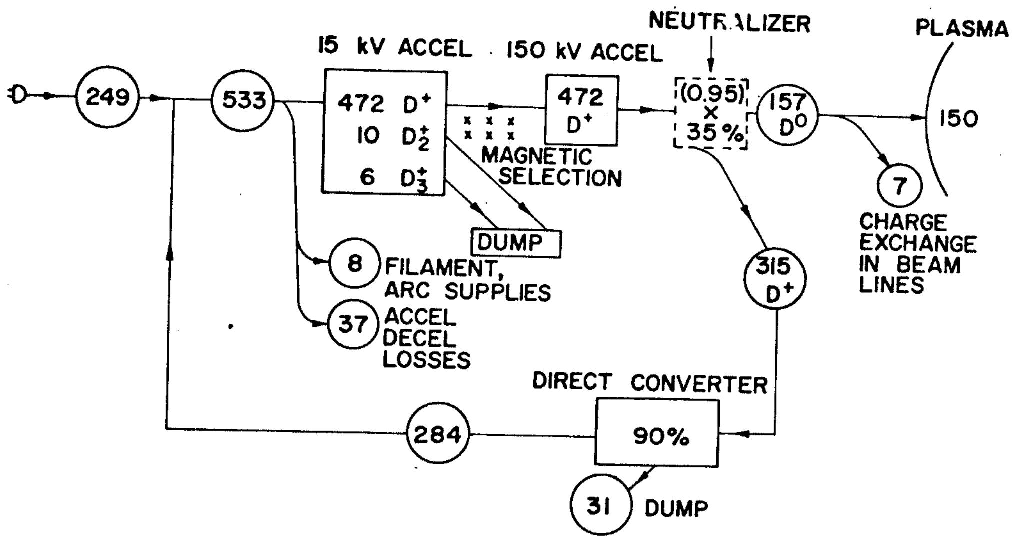
The negative ion system would consist of a low energy (1 keV) positive ion source followed by a cesium vapor cell from which D^- is extracted and then accelerated to the desired energy. In contrast with the positive ion beam, the resulting negative ion beam could be neutralized with good efficiency (approximately 65%) at 150 keV and higher. A negative ion system with direct conversion would require only 195 MW of net electrical power to produce 150 MW of injected power (77% efficiency) at 150 keV and above. Since the neutralization efficiency does not decrease with energy as in the positive ion case, operation at higher beam energy would be possible.

MAGNET DESIGN

The toroidal field magnets consist of 16 large "D" shaped constant tension superconducting coils. The conductor is cryogenically stable and consists of NbTi filaments in an OFHC copper stabilizer embedded in stainless steel structural discs. Each disc is 3.2 cm thick and 55 cm wide, and has 36 conductor turns,

TABLE 5. Parameters of Positive Ion Neutral Beam System - Base Case

Total Injection Power (D°)	150 MW
Energy Beam (D°)	150 keV
Total Injected Current	1000 A equiv.
Number of Beam Lines	12
Sources per Beam Line	20
Total Number of Sources	240
Ion Current per Source (all species)	18.7 A
Total Source Current Density	47 mA/cm^2
Accelerator Power Density	5 kW/cm^2
Beam Path from Neutralizer to Plasma	6 m
Neutral Pressure in Transport Tube	2×10^{-5} Torr
Input Power to Injector System	533 MW
Power Available from Direct Converter	284 MW
Net Power Required	249 MW
Net Injector Efficiency	60%



POWER FLOW (MW) FOR POSITIVE IONS SYSTEM HALF AND THIRD ENERGY COMPONENTS REMOVED AT 15 keV; DIRECT CONVERSION.

FIGURE 5

18 on each side. There are 13 discs in each magnet separated by micarta spacers and clamped together with high strength aluminum bolts. The magnet system has a vertical bore of 10m, a horizontal bore of 6.16m, and a field of 4.2 tesla at a radius of 3.25m, the plasma center. The total energy stored in the magnet is about 6.3 GJ.

A corrected version of the constant tension design by Moses and Young,[18] which accounts for the discreteness of the coils, was used to obtain the "D" shape. The average stress in the conductor is 4700 N/cm^2 which is considerably lower than the yield stress in the copper (8400 N/cm^2).

The TF coils are entirely supported on the central support structure. The shape is that of an ideal constant tension toroidal magnet, without magnetic bending forces anywhere. The gravity load is supported by shear pins located at the top and bottom of the straight leg. The liquid helium dewar will consist of a 3mm thick stainless steel sheet and the vacuum dewar will be made of 6061 T6 aluminum alloy reinforced against the vacuum load by strategically located ribs. There will be 3cm of superinsulation between the walls of the dewar.

A lack of space near the central core makes it difficult to use OFHC copper as a stabilizer and for this reason we offer an alternate design. The excellent electrical and thermal conductivity of high purity aluminum at 4.2K would permit the use of a smaller conductor cross section. However, considerable research and development is needed for the aluminum stabilizer. Space is available for the use of a high purity aluminum stabilizer in high strength aluminum structure, even though stress levels in the 2219 Alloy must be kept lower than for stainless steel. Detailed specifications of both types of TF magnets are listed in Tables 6 and 7.

The pulsed field from the VF coils and the plasma induces ac losses in the superconducting composite of the TF coil. The maximum variation in the field around the coils is $\Delta B \sim 0.5T$ taking place in 1 sec at the edge of the vertical leg of the magnet. The total losses in the TF coils are estimated at about 12.1 kW ad 4.2 K. Shielding the coils with 0.5 cm of high purity aluminum on the inside surface of the liquid helium dewar increases the time constant of the pulsed field to ~25 sec which should prevent the superconductor from going normal.

For simplicity, the VF field coils are water cooled copper operated at 64°C. There are 48 VF coils located inside the bore of the TF magnet. Multiple short length water cooling passages will be needed. The current density in the copper ranges from 1225 A/cm^2 to 1992 A/cm^2 and the time average power loss is 67 MW based on a 75% duty cycle.

The ohmic heating coils will be made of high purity aluminum and operated cryogenically at 4.2 K in order to achieve a high current density. Twenty of the coils will be mounted in the central support structure while the remaining 12 coils will be in individual dewars with separate structure. The estimated power loss for the OH coils is 50 kW which is about 15 MW at room temperature.

BLANKET AND SHIELD

A schematic of the inner and outer blanket and shield used for TETR in the non-testing zones is shown in Fig. 6. The basic structure is 316 SS with large amounts of W, Pb, and B_4C used to slow down and attenuate the neutrons and gamma rays. The blanket and shield is water cooled and some parts of the outer and inner blanket contain water cooled copper VF coils.

The blanket area not used for testing is designed for permanent lifetime of 10-15 years and swing away shields (Figs. 2,4) are included for access to the removable test stations. The breakdown of the specific materials in the blanket and shield is as follows:

	Tonnes
316 SS	792
W	4977
Pb	1735
Cu	157
B_4C	237
	7898

NEUTRONICS

The neutronics analysis of TETR is complicated by the non-circular cross section, toroidal geometry and numerous penetrations for beam ports, vacuum pumps, divertor slots, and test modules. Two dimensional S_n calculations were performed in R-Z toroidal geometry. The details behind the calculations are given elsewhere[19] and Figures 7-8 and Table 8 presents the results in graphical and tabular form. The streaming of neutrons down penetrations was not completed for this paper.

Fig. 7 shows how the neutron flux varies as a function of poloidal angle around the reactor. Note that the flux peaks at ~50° and again at 160° whereas the wall loading peaks at 0° and 180°. The 20-70° position is where the test modules are located. Figure 8 shows how the dpa and helium production rates vary with angle revealing the same behavior as in Fig. 7.

TABLE 6. Specifications of the Torodial Field Coil

On axis field: B_o = 4.2 tesla at R = 3.25m

Maximum field at conductor: B_m = 8.5 tesla at R = 1.6m

Total Energy Stored: 6.3 GJ

Number of Magnets: 16

Superconductor: NbTi

Number of discs: 12

Spacers between discs: 0.5 cm micarta

Insulation between conductor and structure: 0.1cm epoxy fiberglass

Disc Cross Section: 3.2 x 55 cm

	Base Case	Alternate Case
Stabilizer	OFHC copper	High purity Al
Structure	316 stainless steel	2219 Al
Max. design stress (N/m^2)	1.03×10^8	1.54×10^8
Max. design strain	500 μ	0.0019
Conductor current (A)	9838	8050
Average conductor size	2.41 x 1.25 cm	1.41 x 1.25 cm
Number of turns/disc	36	44

TABLE 7. Vertical Field Coil Specifications

Number of Coils	48
Materials used	OFHC Copper
Cooling	Chilled Water
Max. Operating Temp.	65°C (4°C inlet)
Conductor Dimensions (cm)	8.8 x 2.3 (4 cooling holes) 11 x 2.3 (5 cooling holes) 15.4 x 2.3 (7 cooling holes) 17.6 x 2.3 (8 cooling holes)
Current density in the copper	1225 - 1992 A/cm^2
Time avg. power (75% duty factor)	67.2 MW
Cooling water required	21,357 liters/min.
Mass of conductor	144.3 tonnes

Ohmic Heating Coil Specifications

Number of Coils	32
Material used	High conductivity Al
Cooling	Cryogenic pool boiling at 4.2 K
Conductor Dimensions (cm)	10 cm x 1.5 cm 15 cm x 1.5 cm
Maximum current density	3483 A/cm^2
Heat transfer area (1) (2)	600 cm^2/m (10 x 1.5) 1000 cm^2/m (15 x 1.5)
Liquid He boil-off rate	750 liters/min.
Mass of conductor	31.35 tonnes

Figure 6

Schematic of Blanket and Shield Design - TETR

Inner Blanket - (Midplane)

Material	30.7% SS 57.7% Cu 10.1% NbTi 1.6% Insul (TF Coil)	Insulation 1 cm Al 3 cm Mylar 1 cm SS	92% B_4C(a) 4% SS 4% H_2O	92% Pb 4% SS 4% H_2O	96% SS 4% H_2O	92% Pb 4% SS 4% H_2O	92% B_4C(a) 4% SS 4% H_2O	75% Cu 20% H_2O 5% (b)	92% B_4C(a) 4% SS 4% H_2O	92% Pb 4% SS 4% H_2O	92% W 4% SS 4% H_2O	96% SS 4% H_2O	C
Density Factor	1	1	1	1	1	1	1	1	1	1	0.95	1	0.5
Thickness -cm	55	5	5	5	11.5	5	5	7	7.5	7.5	34.5	1.5	0.5

150 cm ← Plasma

Outer Blanket (Midplane)

110 cm

Plasma →

Material	C	96% SS 4% H_2O	92% W 4% SS 4% H_2O	92% Pb 4% SS 4% H_2O	92% B_4C(a) 4% SS 4% H_2O	75% Cu 20% H_2O 5% (b)	92% B_4C 4% SS 4% H_2O	92% Pb 4% SS 4% H_2O	96% SS 4% H_2O	92% B_4C(a) 4% SS 4% H_2O	92% Pb 4% SS 4% H_2O
Density Factor	0.5	1	0.95	1	1	1	1	1	1	1	1
Thickness -cm	0.5	1.5	28	7.5	7.5	30	5	5	15	5	5

(a) 95% TD

(b) Fiberglass Epoxy

The maximum dpa and helium production rates (averaged over a year) are 9.5 and 220 appm respectively. The maximum in the steel damage and He production rates for the carbon curtain is ~10 dpa/yr and ~2500 appm/yr respectively.

It is worth noting that the outer blanket and shield provides a 5×10^6 reduction in dpa values to the TF coils and the inner blanket and shield provides a 2×10^5 fold reduction. The resulting maximum dpa rates in the Cu stabilizer (4×10^{-5} dpa/yr) mean that the magnets will have to be warmed to room temperature every few years to remove 90% of the displacement damage.

HEAT TRANSFER AND COOLANT CONSIDERATIONS

As stated previously, the main coolant of the blanket and shield is water while helium is used to cool the test modules which will run at higher temperatures (300-600°C). The distribution of the 346 MW of energy emanating from the plasma is as follows.

150 MW in divertor area
157 MW in the permanent blanket
39 MW to the test modules
346 MW

The coolant conditions for the permanent blanket are listed in Table 9 and reveal an inlet temperature of 150°C and an outlet temperature of ~225°C. The maximum structure temperatures are also listed in Table 9 and show that the maximum steel temperature is ~250°C and the maximum temperature for W, Cu, Pb and B_4C are 400°C, 70°C, 225°C and 400°C respectively.

TABLE 8. Summary of Selected Neutronics Parameters - TETR(a)

Neutron Prod. Rate During Burn (inst.)	$9.2 \times 10^{19} s^{-1}$
Max. Neutron Flux - (inst.)	
(>13.5MeV)	$9 \times 10^{13} cm^{-2} s^{-1}$
(>0.1 MeV)	$2.5 \times 10^{14} cm^{-2} s^{-1}$
(>0 MeV)	$4 \times 10^{14} cm^{-2} s^{-1}$
Max. Dpa Rate-Carbon Curtain	$8.7\ y^{-1}$
Max. Dpa Rate-316 SS	$9.5\ y^{-1}$
Max. He Prod.-C	$2500\ y^{-1}$
Max. He. Prod.-316 SS	$220\ y^{-1}$
Max. Dpa Rate-VF Coils	$8 \times 10^{-3} y^{-1}$
Max. Dpa Rate - TF Coils	$4 \times 10^{-5} y^{-1}$

(a) 70% PF except where noted.

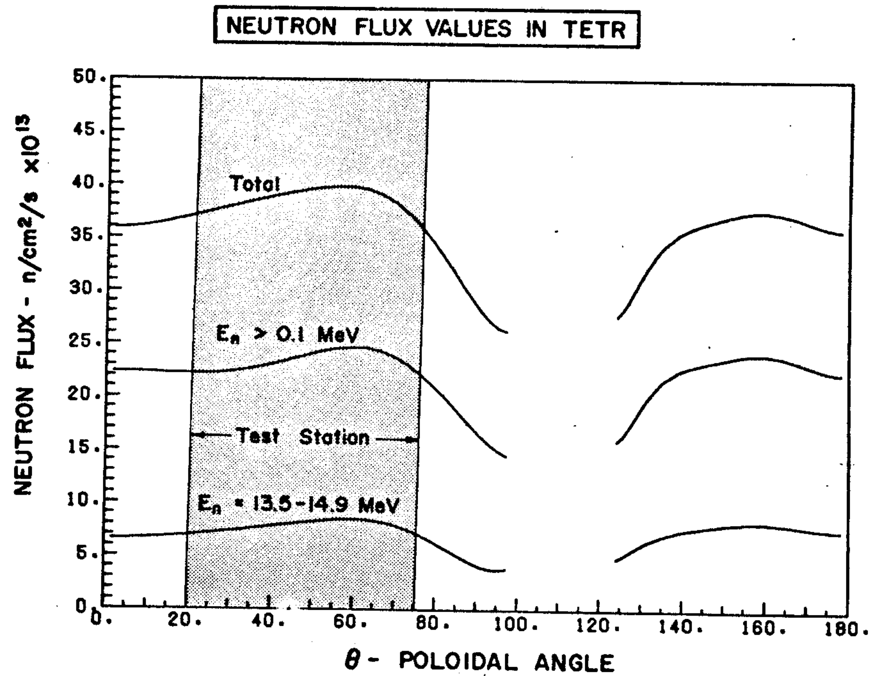


FIGURE 7

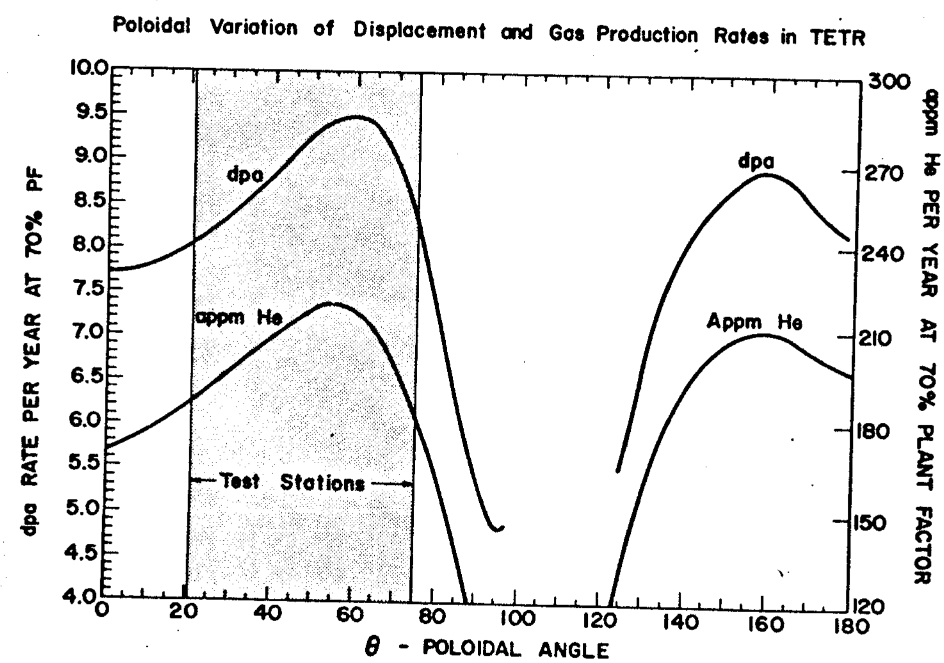


FIGURE 8

The heat from the divertor collector, blanket and shield, as well as 100 MW_t collected from the beams outside the reactor is all deposited in the wet cooling towers. A total water requirements of the plant is 6.2 x $10^4 m^3$/day.

The temperature of the carbon curtain is calculated from the particle and photon heating rate of 5 watts/cm^2 in addition to 10 watt/cm^3 neutron heating rate. The resulting equilibrium temperature for the 60 second burn is 1300°C and this drops to ~900°C and the 12.5 second down time between burns.

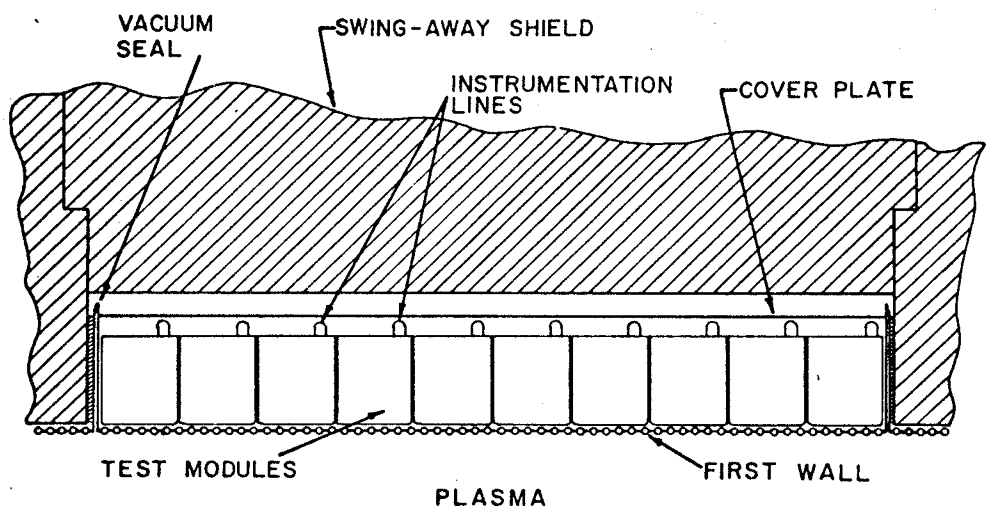
TEST STATIONS

There are 16 test stations whose functions are listed in Table 10. The first seven stations (7 x $10^5 cm^3$ actual test area) are devoted to testing the primary alloy and two stations (2 x $10^5 cm^3$) are devoted to testing a backup alloy. There is one stationeach devoted to testing carbon, refractory metals, and advanced materials. There are two stations devoted to testing coolant-material-magnetic field effects and one station devoted to neutron multipliers and solid breeder materials. Finally, there is one station devoted to testing methods for locating defects in the first wall of fusion reactors, a problem which has received very little attention so far, but one which will probably be critical for power reactors.

TABLE 9. Summary of Heat Transfer Parameters - TETR

Main Coolant-Blanket and Shield	H_2O
Inlet Temperature °C	150
Out Temperature °C	225
Pressure-Outlet (MPa)	2.6
Test Cell Coolant	He
Inlet Temperature °C	200
Outlet Temperature °C	500
Pressure (MPa)	5.2
Particle Collection Coolant	H_2O
Inlet Temperature °C	75
Outlet Temperature °C	150
Pressure (MPa)	2.8
Maximum 316 SS Temp.	250
Maximum W Temp.	400
Maximum Cu (VF Coil) Temp.	150
Maximum B_4C Temp. °C	225
Maximum C Curtain Temp.	1300

FIGURE 9



CROSS SECTION VIEW OF A TYPICAL TEST STATION

TABLE 10. TETR Test Program

Module #	Material	In Situ	Post Irradiation
1,2	Primary Demo Alloy	Fatigue	Swelling & Mech. Prop.
3,4	Alloy	Creep	"
5,6	Alloy	(Crack Prop. & Biaxial)	"
7	Alloy	Surface Effects	"
8	Carbon	Plasma Int.	Growth and Mech. Prop.
9	Steel-Li	Fatigue	Corrosion
10		Dynamic Testing of Defected Blanket Designs	
11		Non Li Coolants	
12		T_2 Breeder and n Multipliers	
13,14	Backup Demo Alloy	Fatigue Creep Biaxial	Swelling & Mech. Prop.
15	Refractory Metals	Fatigue Creep	Swelling & Mech. Prop.
16	Advanced Materials		

A schematic of one of the test station is given in Fig. 9 which shows the first wall, the 10 cm test zone and some of the 20 individual test modules, each devoted to various in situ testing. The proposed testing matrix for the primary alloy is given in Table 11. The matrix proposes over 5000 in situ tests that would be performed in 7 test stations and provide information on creep fatigue, crack propogation, etc. This information would be forth coming as shown in Figure 11. Noting the fact that the information at 5 different temperatures and under a variety of stress-strain, metallurgical and environmental conditions would be obtained simultaneously, we see that by 1988 data up to 10 dpa and 220 appm helium levels would be obtained. By the 1992 DPR final decision date (Title 1 Construction begins), data on fatigue, creep, biaxial stress and crack propagation will be available for up to ~50 dpa and 1000 appm helium. It should be noted that this information is obtained under typical CTR dpa rates and with the appropriate PKA spectra in contrast to fission reactor or spallation neutron sources.

11. Tritium and Radioactivity in the TETR Structure

A summary of the tritium parameters in TETR is given in Table 12 and this shows that .1419 g of tritium are injected per second of operation. At the low burn fraction of the machine, each tritium atom has to be handled several hundred times.

Since each tritium atom must be recycled many times and each handling involves the separation of it from other exhausted fuel components, fuel recycle is handled by a separation system which first separates fueling species (deuterium and tritium) from impurity argon, nitrogen, oxygen and hydrocarbons by means of a

TABLE 11. In Situ Demo Structural Materials Test Matrix

Property	Base & Weld Metal	Alloy Variation	Stress Strain Range	Backup Material & Redundancy	Envir.	Temp.*	Total
Fatigue	2	4	5	4	3	5	(2400)
Crack Growth	2	4	5	4	2	5	(1600)
Creep	2	4	5	4	2	5	(960)
Biaxial	1	4	4	4	1	5	(320)
							5280

*Irradiation Temperature: 200, 300, 400, 450, 500°C

Pd-Ag diffuser. Then the fueling species are cryogenically distilled into two streams consisting of (1) 90% D_2, 10% DT and (2) 90% T_2, and 10% DT for the neutral beam injectors and pellet maker, respectively.

From the injected power and beam energy and accounting for a 60% beam efficiency, the required deuterium atom flow is .017 moles deuterium atoms/sec into the neutral beam injectors.

The tritium atom flow into the plasma calculated from the plasma particle density, particle confinement time and plasma volume is .047 moles T atoms/sec.

The tritium inventory is listed also in Table 12 and it is shown that with a 5 hour recycle time on the cryopumps, the inventory in the vacuum system is 2540g while another 740 grams is held up in the distillation and pellet fabrication area. Assuming that at least a 1 day reserve of T_2 will be required (amounting to 12180 grams) the total inventory is 15460g.

From the thermonuclear power, the tritium burnup is calculated to be 34g per day or 12.4 kg per year. At a decay rate of 6% per year, decay losses are calculated to be 927g per year.

Table 12. Summary of Tritium Parameters in TETR

Feed rate g/day	12180
Burnup Rate g/day	34
Inventory (g)	
Cryopumps	2540
Distillation	740
Storage (1 day)	12180
Total	15460g
Release Rate Per Day	10Ci

At the low 1st wall temperature ($200^{\circ}C$), tritium leaking from the plasma into the water coolant is small, about .09 Ci/day or 33 Ci/yr. Leakage into the heat exchanger will be smaller at the lower temperatures involved. Leakage into the coolant of the 6 test modules is small, about .0065 Ci per day or 2.4 Ci per year. For TETR, the release rate of tritium to the environment is estimated to be 10 Ci/day.

The induced radioactivity in the blanket and shield, and TF coils is plotted in Fig. 11 for both buildup and shutdown. Note that the blanket and shield approach the 1-5 curie per watt level in a few hours while it takes over 10 years to decay to 10^{-2} curie/watt. Even the TF coils have accummulated a significant level of activity in a few months.

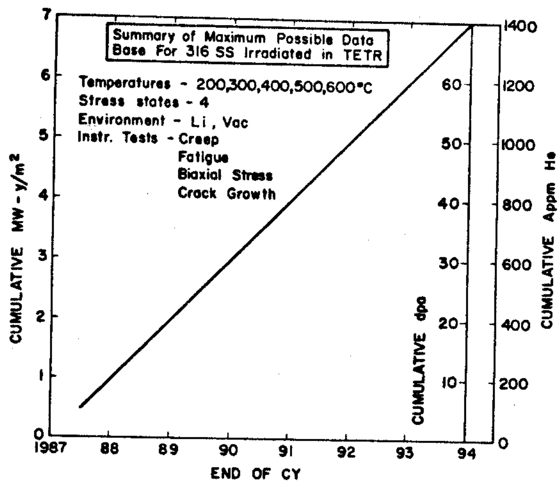


FIGURE 10

Table 13. Selected Power Requirements and Major Operating Cost Items

Power (a)		10^6 \$ (b)
Neutral Beams	249 MWe	15.3
VF Coils	60	3.7
OH Coils	30	1.8
Refrigeration	3	0.2
Misc.	12	1.0
	354	22
Tritium Costs (c)		
Burnup (Per Year)		16.9
Decay (12.2 kg Inventory)		3.5
		19.4

a) averaged over burn cycle
b) assuming 10 mills/kWh
c) \$5000/g

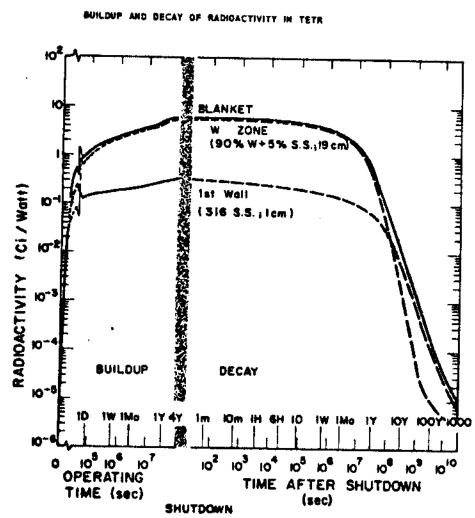


FIGURE 11

12. Selected Requirements and Costs

A listing of the major power demands is given in Table 13 showing that a time averaged value of 354 MWe. This is, of course, dominated by 249 MWe required by the beams to deliver 150 MW to the plasma. The normal VF coils also need 60 MWe and the OH coils require an additional 30 MW. If one uses as current cost of power at 10 mills/kWh (northwest USA), the total power costs are approximately $22 M per year.

The cost of buying tritium fuel at 5000 $/g for TETR is calculated to be about $17 M per year while an additional $3.5 M must be allocated to replace T_2 which decays in the approximately 12.2 kg inventory. Hence, it is clear that the total operating costs of TETR will be less than $75 M per year even using liberal operating and maintenance costs but excluding the cost of the experimental program.

13. Conclusions

Our preliminary investigation has revealed the following conclusions -

·It should be possible with 1980-5 state-of-the-art plasma physics and engineering technology to produce an integrated neutron wall loading of ~1 MW-yr/m^2 per year of operation on meaningfully instrumented samples subjected to the characteristic environment of a tokamak reactor.

·The required magnet and beam technology to confine and heat a TCT plasma sufficiently should be readily available by 1985.

·An irradiation facility like TETR can be a complete power and tritium consumer and still be operated for less than $75 M per year.

·By the use of water cooling, periodic annealing and/or the use of a carbon ISSEC, one should be able to attain a structural wall life of ~15 MW-yr/m^2 in stainless steel operating at 250 °C and thus greatly reduce down time and the generation of radioactive wastes associated with first wall replacements.

·The materials test information to be provided by the TETR can meet even the most stringent requirements on DPR materials by the time a decision would have to be made if a TETR were operating, as envisioned here, by 1987.

·The TETR can provide valuable information about other engineering materials such as backup and advanced alloys, solid breeders and neutron multipliers, various coolant material interactions and methods for detecting leaks in power reactors.

·The large test volume of TETR will produce, by 1990, more specimens tested in situ at the required dpa and transmutation levels than any other simulation test facility thus far proposed.

Acknowledgements

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