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Oak Ridge TNS Program: System Description Manual

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OAK RIDGE NATIONAL LABORATORY
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ABSTRACT

This document provides a systems description of the Reference Design for The Next Step (TNS) evolved at Oak Ridge National Laboratory (ORNL) during FY 1978. The description is presented on the basis of 24 individual device and facility systems. Additional information on these systems, the Reference Design, and the FY 1978 Oak Ridge TNS activities can be found in the associated technical memoranda, ORNL/TM-6720 and ORNL/TM-6722–ORNL/TM-6733.

INTRODUCTION

The Next Step (TNS) represents that phase of fusion energy development in which the major emphasis would be directed towards engineering testing and demonstration. The objective of the TNS studies (initiated by the Department of Energy's Office of Fusion Energy) has been to define the characteristics and requirements of a facility dedicated to the engineering testing phase of fusion power development. For this reason, the TNS study results are providing a basis for defining an Engineering Test Facility (ETF). Because the scientific basis required for a TNS/ETF will first be available for the tokamak concept, the reactor core of the facility has been based on the tokamak concept. However, the commitment to an ETF with a reactor core based on the tokamak concept does not represent a commitment to tokamaks as the ultimate power reactor concept.

The TNS studies at Oak Ridge National Laboratory (ORNL) were initiated in FY 1977. During FY 1977 the Oak Ridge effort pursued scoping studies in three broad areas: plasma engineering, systems modeling, and program planning. Based upon the findings of the FY 1977 efforts, it was judged that continued activities in the Oak Ridge TNS program should be directed towards a preconceptual design with particular emphasis placed on engineering feasibility. As a point of departure for the FY 1978 activities, a Baseline Design was selected, based on the systems modeling effort of FY 1977. The primary objective of our FY 1978 TNS effort has been to evolve the Baseline Design towards a preconceptual design. However, it is emphasized that the FY 1978 effort was not intended to lead to a completed preconceptual design. Therefore, the design resulting from this year's effort is referred to as a Reference Design, rather than as a preconceptual design.

This document provides a general description of the Reference Design on the basis of 24 individual device and facility systems taken from the Work Breakdown Structure (WBS) of Ref. 1. Most system descriptions include the function of a particular system, its design requirements, a general description of that system, and a table listing pertinent design parameters (e.g., geometric, electromagnetic, thermal, and nuclear) and their values.

The primary thrust of this year's TNS activities was in five areas: systems engineering, mechanical design (vacuum topology, assembly, and maintenance), poloidal field system, plasma dynamics, and particle control. Systems associated with these five areas are consequently described in greater depth and with more definition than systems unrelated to these five areas. In the latter case, the systems have been adapted and modified from last year's TNS studies.[2]

REFERENCES

1. D. Steiner, T. G. Brown, Y-K. M. Peng, R. L. Reid, M. Roberts, T. E. Shannon, and P. T. Spampinato, *Oak Ridge TNS Program: Context, Scope, and Baseline Design of the FY 1978 Activities*, ORNL/TM-6201, Oak Ridge, Tennessee (May 1978).
2. TNS Engineering Staff, *Four Ignition TNS Tokamak Reactor Systems — Design Summary*, WFPS-TME-071, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania (October 1977).

1. SUMMARY

The Reference Design tokamak is an ignited, air core device burning deuterium-tritium (D-T). The duty cycle is 89% with a steady-state burn of 500 sec. The plasma radius is 1.2 m, the major radius is 5.0 m, and the plasma elongation is 1.6. The fusion power density is 5.0 MW/m^3, which results in 1140 MW of fusion power during the burn. Exothermic reactions in the first wall and shield are assumed to increase the thermal power to 1450 MW (based on an energy release of 22.4 MeV per fusion event).

A water-cooled, tubular first wall in conjunction with a bundle divertor removes the energy associated with the fusion alpha particles. Both the first wall and the bundle divertor are designed to remove all the alpha particle energy; however, the distribution of this energy to each component is estimated to be approximately 50%. A stainless steel shield, cooled by borated water, absorbs the bulk of the energy associated with the fusion neutrons. All the thermal energy, including that deposited on the first wall and the divertor, is dumped to a cooling tower.

The stainless steel toroidal vessel is composed of 16 bolted segments, and the entire tokamak is situated within an evacuated containment building. A poloidal field (PF) system, consisting of a superconducting ohmic heating (OH) solenoid and both superconducting and copper equilibrium field (EF) coils, induces the plasma current and provides plasma position control. Plasma startup is assisted by the injection of 1 MW of rf power at a frequency of 120 GHz. Bulk heating is accomplished by the injection of 50 MW of 150-keV neutral beams into the plasma. The confining toroidal field is provided by 12 Nb_3Sn superconducting coils (TF coils). Plan and elevation views of the Reference Design are shown in Figs. 1.1 and 1.2, respectively.

The Reference Design operating cycle is shown in Fig. 1.3. The mission objective for the Reference Design covers a 10-year period of operation. Four phases of operation are included in this 10-year period:

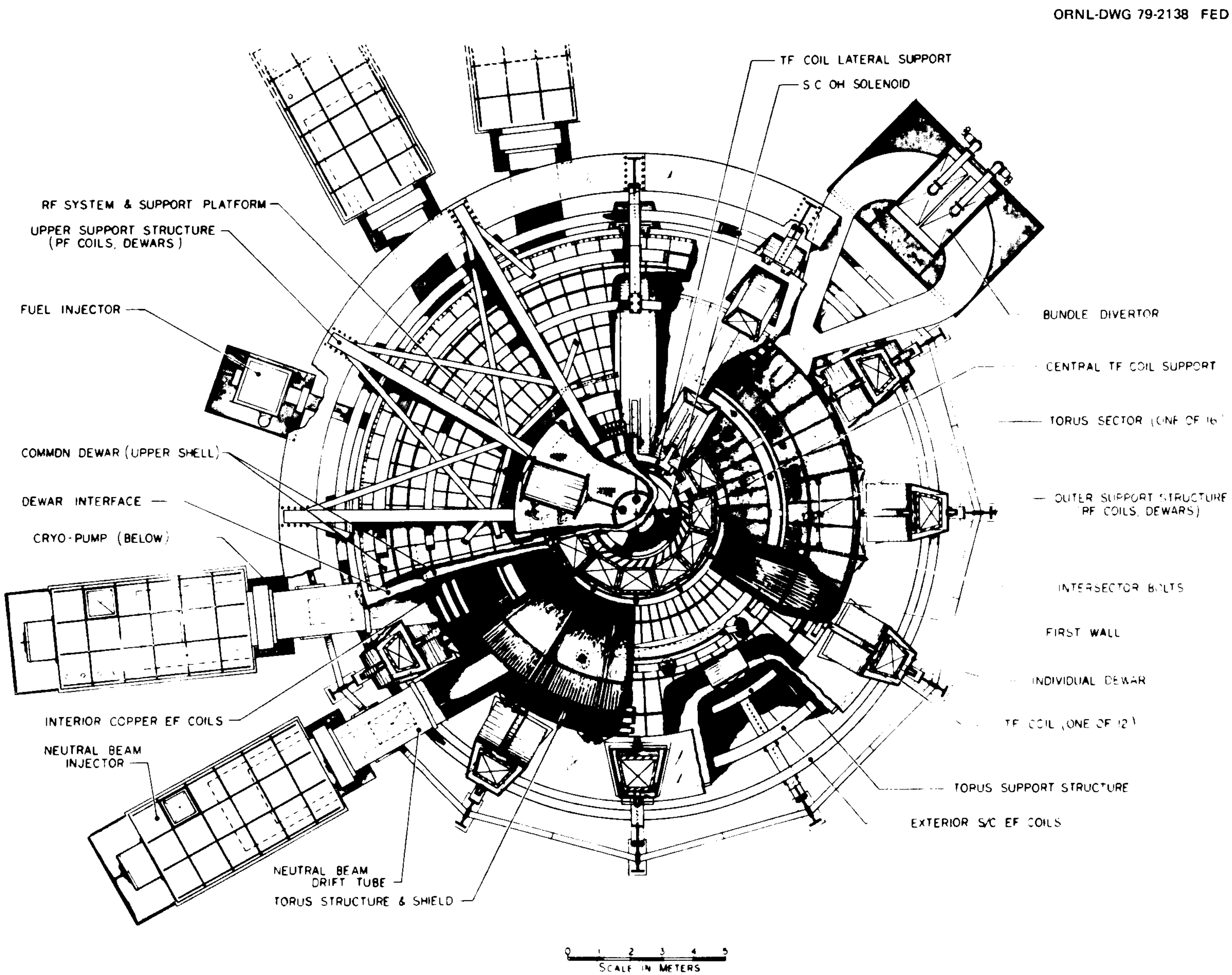

Fig. 1.1. Plan view of Reference Design.

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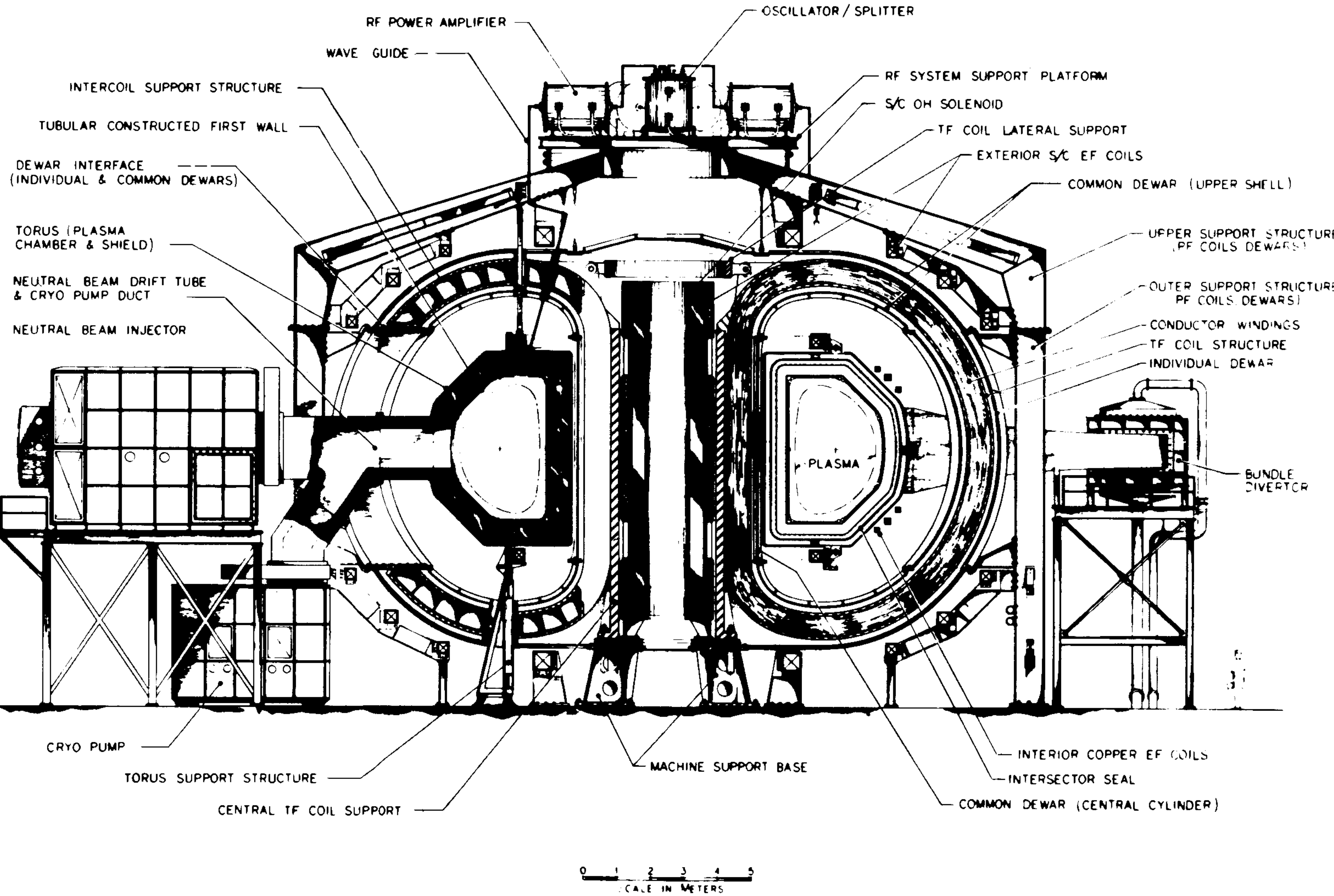

Fig. 1.2. Elevation view of Reference Design.

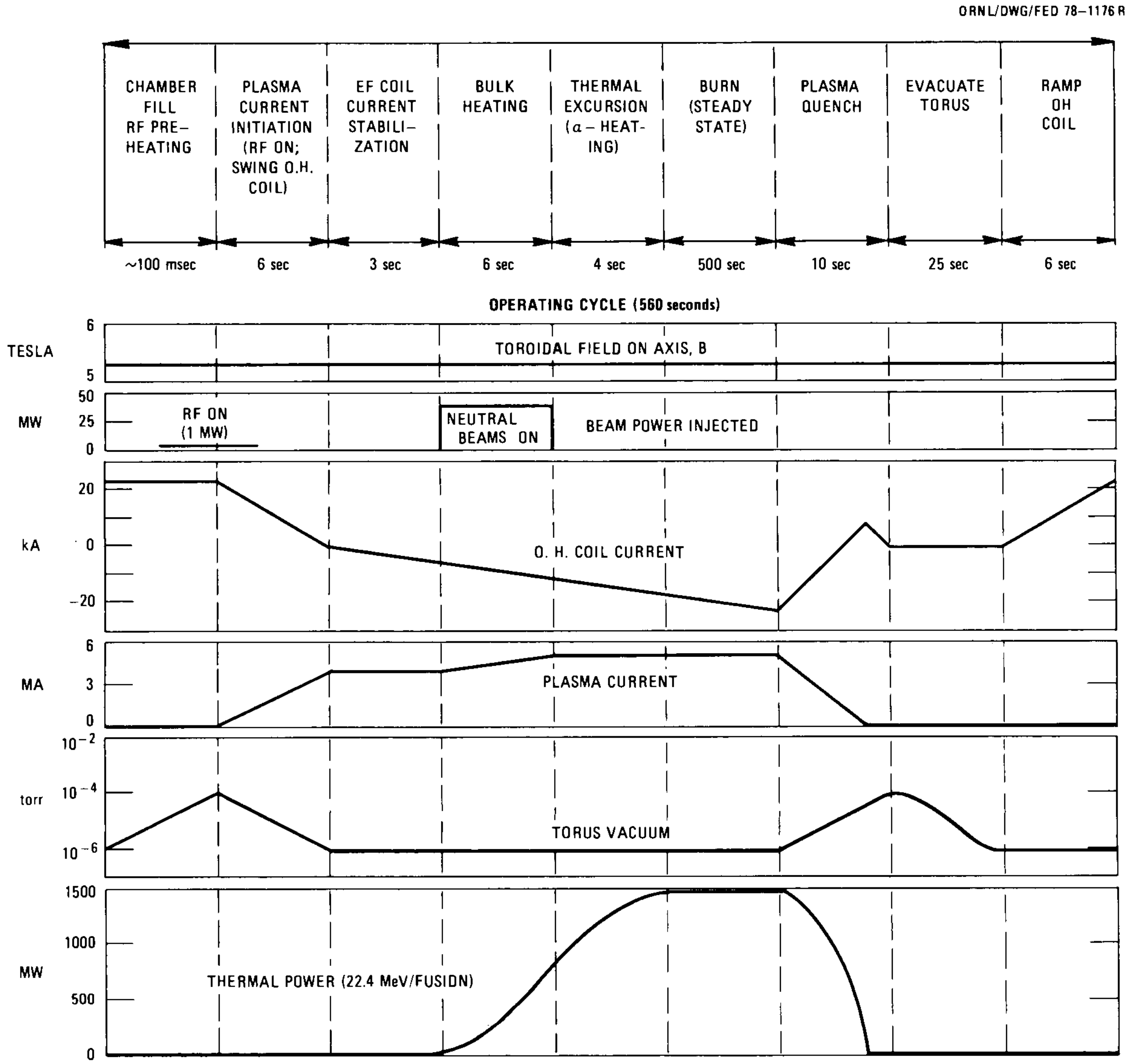

Fig. 1.3. Reference Design operating cycle.

Phase I. Systems integration and checkout (0.5 year),

Phase II. Hydrogen operation, including long pulse (1.5 years),

Phase III. Ignition testing (2.5 years),

Phase IV. Technology and engineering test (5.5 years).

Portions of each phase will operate at a reduced pulse rate, i.e., less than the design value of approximately 50 shots per 8-hr shift. The total number of cycles for the Reference Design is estimated to be 450,000 over the 10-year period of operation.

A summary of the parameters for the Reference Design is given in Table 1.1. Table 1.2 provides a brief system description of the Reference Design.

Table 1.1. List of parameters for Reference Design

Geometric	
Plasma radius, a	1.2 m
Plasma elongation, σ	1.6
Aspect ratio, A	4.17
Major radius, R	5.0 m
Plasma edge-to-winding distance, Δ	1.28 m
Plasma	
Beta, $\overline{\beta}$	7.0%
Plasma temperature, $\overline{T}$	12 keV
Ion density, $\overline{n}$	2×10^{20} m^{-3}
Impurity indicator, Z_{eff}	<1.5
Safety factor, q	3.8
Plasma current, I_p	5.0 MA
Burntime (steady-state), τ_B	500 sec
Duty cycle, dc	89%
Cycle duration	560 sec
Energy confinement time, τ_E	1.2 sec
Pellet fueling rate	4.6×10^{22} particles/sec
Beam power into plasma, P_{inj}	50 MW
Beam energy, E_{inj}	150 keV
Microwave power, P_{rf}	1 MW
Microwave frequency, f	120 GHz
Plasma volume, V_p	227 m^3
Fusion power density, P_d	5.0 MW m^{-3}
Power (fusion), P_f	1140 MW
Total power (22.4 MeV/event[a]), P_{tot}	1450 MW

Table 1.1 (continued)

Electromagnetic	
Toroidal field on axis, B_T	5.3 T
Maximum toroidal field at coil, B_m	10.9 T
Number of TF coils	12
Magnetic field ripple at plasma edge, δ_a	1.5% peak to average
TF coil type	Superconducting (Nb_3Sn)
OH solenoid type	Superconducting (NbTi)
PF coil type	
Inside TF bore	Copper
Outside TF bore	Superconducting
Total volt-seconds, ϕ_{vs}	83
Magnetic field in OH solenoid, B_{OH}	8.0 T
Thermal	
First wall coolant	Water
Shield coolant	Borated water
Refrigeration load (TF coils), P_{ref}	48 kW at 5K
Power to first wall, $P_{\alpha/2}$	113 MW
Power to divertor, $P_{\alpha/2}$	113 MW
Power to shield, P_s	1224 MW
Plasma edge particle loss rate	3×10^{23} particles/sec
Nuclear	
Neutron wall loading, L_w	2.4 MW/m^2

[a] 22.4 MeV/event assumed.

Table 1.2. Summary system description

Plasma	Expected to achieve an average power density of 5 MW/m^3 and operate with $\bar{\beta} = 7\%$. Temperature and density during steady-state burn are taken to be 12 keV and $2 \times 10^{20}\ m^{-3}$, respectively. Plasma impurity control is provided by a bundle divertor. A 500-sec burn out of a 560-sec tokamak operating cycle yields a duty factor of 89%. Plasma radius is 1.2 m with an elongation of 1.6 (D-shape); major radius is 5.0 m. An average neutron wall loading of 2.4 MW/m^2 results.
Limiter	Two poloidal graphite limiters, assumed to be passively cooled, are used to protect the first wall.
First wall	The first wall and divertor are each designed to handle all the energy associated with the alpha particles. However, the energy to the first wall is estimated to be only 50% of the alpha particle energy. The water-cooled first wall is composed of stainless steel tubes and is divided into 96 modular panels.
Shield	Composed of stainless steel balls and borated water and divided into 16 removable modules. Sized for adequate attenuation of neutrons in order to protect the superconducting coils.
Machine structure	Includes the bucking cylinder, all TF coil supports, PF coil supports, dewar structure, and torus plasma chamber; designed to react all operational mechanical loads as well as seismic load conditions; constructed of austenitic stainless steel. The intercoil support structure for the TF coil is designed for a single coil fault condition; all TF coil support structure is cryogenically cooled.
Poloidal field system	Consists of a superconducting OH solenoid, normal copper EF coils located inside the TF coil bore, and superconducting EF coils located outside the TF coil bore. Will supply 83 V-sec while operating with a half-biased OH system swinging from +8 T to -8 T, adequate for startup and a 500-sec burn.

Table 1.2 (continued)

Toroidal field system	Composed of 12 forced-flow superconducting (Nb_3Sn) coils which produce a field on axis of 5.3 T and a maximum field at the coil of 10.9 T. Sized for a magnetic field ripple of 1.5% at the plasma edge.
Dewars	Mechanically joined dewars enclose the superconducting coils, the central supporting column, and the intercoil support members and provide a vacuum barrier to reduce convection heat loads and a cryogenically cooled thermal shield to reduce radiation heat loads.
Divertor	The divertor and first wall are each designed to handle all the energy associated with the alpha particles. However, the actual energy to the divertor is estimated to be only 50% of the alpha particle energy. The bundle divertor transports charged particles away from the first wall by diverting magnetic field lines outside the plasma chamber. Flowing lithium is used to collect the diverted particles and absorb their energy.
Bulk heating	Four beam lines with three positive ion sources per line inject 50 MW of 150-keV deuterium neutral beams into the plasma for up to 6 sec to achieve required plasma temperatures. A direct recovery system to collect the energy of the unneutralized particles is a component of the beam line. An overall beam line efficiency of approximately 50% is estimated.
rf heating	An rf assisted startup is accomplished by injecting 1 MW of 120-GHz microwaves into the plasma from the high field side for approximately 2 sec during startup. Five injectors are used.
Electrical energy storage	Provided by two motor-generator flywheel (MGF) sets with a combined rated capacity of 600 MVA and a deliverable energy of 4.3 GJ over 35 sec. Electrical energy during most of the burn is supplied directly from the utility grid.

Table 1.2 (continued)

Electrical energy conversion	Receives ac power from the motor-generator flywheel sets and converts it to dc for each coil set. Transformers and solid state rectifiers are used.
Electrical distribution	The primary system receives ac power from the utility grid at 138 kV and transforms 600 MVA to a 13.8-kV load bus. The secondary system transfers power from the electrical energy conversion system to the coil sets.
Emergency standby power	Consists of batteries that store sufficient energy to controllably terminate the plasma discharge in the event of loss of power from the utility grid or the primary MGF sets.
Torus vacuum pumping	During initial pumpdown and after plasma burn cycles, accomplished by mechanical vacuum pumps and cryosorption pumps. During the burn, provided primarily by the divertor system.
Heat transport	Water cooling systems remove heat from the first wall, shield, copper coils, and portions of the neutral beam lines. Cryogenic cooling systems remove heat from the superconducting coils and high vacuum cryosorption pumps. The heat removed by these systems is rejected to the environment by a water cooling tower.
Fuel injector	The fuel injector delivers solid D-T pellets to the plasma at velocities of 1000-2000 m/sec.
Fuel handling	Provides for removing tritium, deuterium, and helium from the spent plasma, the divertor, and the vacuum pumps. Processing and purification are accomplished by combinations of mechanical filtration, gettering, cryogenic trapping, cryogenic distillation, permeation-diffusion, and sorption.

Table 1.2 (continued)

Radwaste	Will consist of portions of the tokamak itself and those components which become contaminated through contact with the coolant. Disposal is expected to be performed using techniques established in fission programs.
Ventilation	Major functions include containment of activated air and tritium leaks. The use of a vacuum building diminishes the activated air problem and is compatible with a tritium cleanup system.
Instrumentation, control, and data handling	Automatically controls the tokamak plasma and systems through all phases of operation. Controls are centralized and are under the command of one operation station.
Assembly and maintenance	The torus is divided into 16 mechanically assembled sectors, each replaceable as a unit with integrally mounted portions of the first wall and shield. Bolted rather than welded assembly is possible since requirements for vacuum tightness are relaxed by the use of an evacuated reactor building. The TF coil bore is enlarged to allow relocation of PF coils to a parking position during sector replacement. Disassembly is done remotely.
Facilities	The tokamak containment building is an evacuated structure with base pressure of 10^{-4} torr. The remaining facilities are conventional atmospheric buildings.

2. PLASMA DESCRIPTION

The Reference Design plasma is assumed to operate in an ignited mode. The parameters listed in Table 2.1 represent best-estimate extrapolations based on our current understanding of plasma physics. It is emphasized that these parameters are to be viewed as plausible, not definitive. The Reference Design plasma is briefly described in terms of the startup and burn phases of the operating cycle.

Table 2.1. Reference design plasma parameters

Geometric	
Plasma radius, a	1.2 m
Plasma elongation (D-shape), σ	1.6
Aspect ratio, A	4.17
Plasma volume, V_p	227 m^3
Field on axis, B_T	5.3 T
Fusion startup	
rf power (preheating), P_{rf}	1 MW
rf frequency, f	120 GHz
rf injection duration, τ_{rf}	2 sec
Volt-seconds required for current buildup, ϕ_{vs}	53 V-sec
Full plasma current at low beta, I_p	4.0 MA
Neutral beam injected power, P_{inj}	50 MW
Neutral beam energy, E_{inj}	150 keV
Neutral beam injection duration, τ_{inj}	$\leq$6 sec
Beam injection angle from perpendicular, θ	>16° in the direction of plasma current
Burn	
Power density, P_d	5 MW/m^3
Fusion power, P_f	1140 MW
Total power, P_{tot}	1450 MW

Table 2.1 (continued)

Burn	
Alpha power, P_α	225 MW
Neutron wall loading, L_w	2.4 MW/m^2
Burntime (steady-state), τ_B	500 sec
Cycle time, τ_c	560 sec
Duty cycle, dc	89%
Volt-seconds required for heatup and burn, ϕ_{VS}	30
Plasma current at high beta, I_p	5.0 MA
Plasma temperature, $\overline{T}$	12 keV
Plasma density, $\overline{n}$	2×10^{20} m^{-3}
Volume beta, $\overline{\beta}$	7.0%
Volume-averaged energy confinement time, $\overline{\tau}_E$	1.2 sec
Plasma edge particle loss rate	3×10^{23}/sec
Plasma fraction recycled at edge as neutrals	85%
Pellet fueling rate	4.6×10^{22} pellets/sec
Impurity indicator, Z_{eff}	<1.5
Safety factor, q	3.8
TF ripple at plasma edge, δ_a	1.5%
TF ripple at plasma center, δ_o	0.1%

2.1 STARTUP

The process of startup has three phases: plasma initiation, current buildup, and heating to ignition. The desired level of prefill gas pressure is $\sim 10^{-4}$ torr. The initiation phase assumes rf ($\sim$1 MW at 120 GHz) gas breakdown and electron preheating to a few hundred electron volts at the upper hybrid resonance layer in a time scale of 0.1 sec. Loop voltage is then applied to establish an electrical current in the toroidal direction while the rf power is maintained. It is assumed that the plasma impurity and the current channel are well controlled during this phase. This, together with successful preheating, should minimize the resistive volt-seconds and loop voltage during current startup. The full plasma current ($\sim$5 MA) is sufficient to contain fusion alpha particles.

A bundle divertor system is assumed to channel runaway electrons away from the first wall, reduce impurity production close to the plasma, shield the plasma from impurities originating at the walls, and allow for good particle control (density control) of the plasma during the heating phase. For the Reference Design, the full bore plasma is heated to ignition by the injection of 50 MW of 150-keV neutral beams for approximately 6 sec. During injection heating, the plasma density is increased from 5×10^{19} m^{-3} to 2×10^{20} m^{-3} to allow for initial beam heating at the plasma center. The plasma can ignite near the center when the volume-averaged beta exceeds $\sim$4% and $\bar{n}$ approaches $\sim 10^{20}$ m^{-3}.

2.2 BURN

Fusion alpha particle heating brings the plasma beta up to around 7% before pressure-driven MHD instabilities set in to limit its further increase. The burning plasma can have a relatively wide range of density, temperature, and power variations; we choose 2×10^{20} m^{-3}, 12 keV, and 1140 MW, respectively, as representative values.

The bundle divertor handles most of the particle loss and part of the energy loss from the plasma and provides for impurity control. Fueling to sustain the burn is accomplished by pellet injection. An upper bound of the power and particle handling capability requirements can be estimated by assuming that thermal power leaves the plasma solely by particle convection and edge recycling, given 3×10^{23} particles/sec at 4.7 keV. This leads to a particle edge recycling time of around 0.15 sec. The recycling temperature could be reduced by increasing the recycling particles from the divertor to the plasma or by enhancing the energy transport to the first wall; the net effect is to increase the particle flux and the heat load to the first wall. It is assumed that no more than half the total thermal power goes through the divertor chamber.

The Reference Design assumes partial pellet penetration at a fueling rate of 4.6×10^{22} particles/sec and an average confinement time for the pellet-fueled plasma particles of approximately 1 sec. It is further assumed that 85% of the plasma particles lost at the plasma edge recycle as neutral gas.

The plasma inductive volt-seconds at I_p = 4 MA is estimated to be around 51 Wb. Plasma heating by rf power before and during current startup is assumed to limit the resistive loss of volt-seconds to 2 Wb. Before ignition, the plasma loop voltage is estimated to be 0.4-0.5 V at I_p = 4-5 MA. During flux conserving heating of the plasma to $\overline{\beta}$ = 7%, the increased vertical field inductively drives the plasma current to nearly 5 MA. During the burn, the plasma loop voltage is estimated to be near 0.05 V. With a plasma heatup time of 10 sec and a burntime of 500 sec, an additional primary flux swing of 30 Wb is needed, leading to a total requirement of 83 Wb.

3. LIMITER SYSTEM

3.1 FUNCTION

Limiters provide physical protection to the first wall during uncontrolled plasma startup or shutdown. In addition, they can provide plasma shaping as a scrapeoff mechanism. Limiters are either fixed or movable and either actively or passively cooled.

3.2 DESIGN DESCRIPTION

The limiter system is made up of two sets of poloidal limiters, located 180° apart. Each set consists of four fixed units, as shown in Fig. 3.1. Due to the irregular shape of the torus cross section and the fact that the limiters are for first wall protection only, and not plasma shaping, fixed units were chosen.

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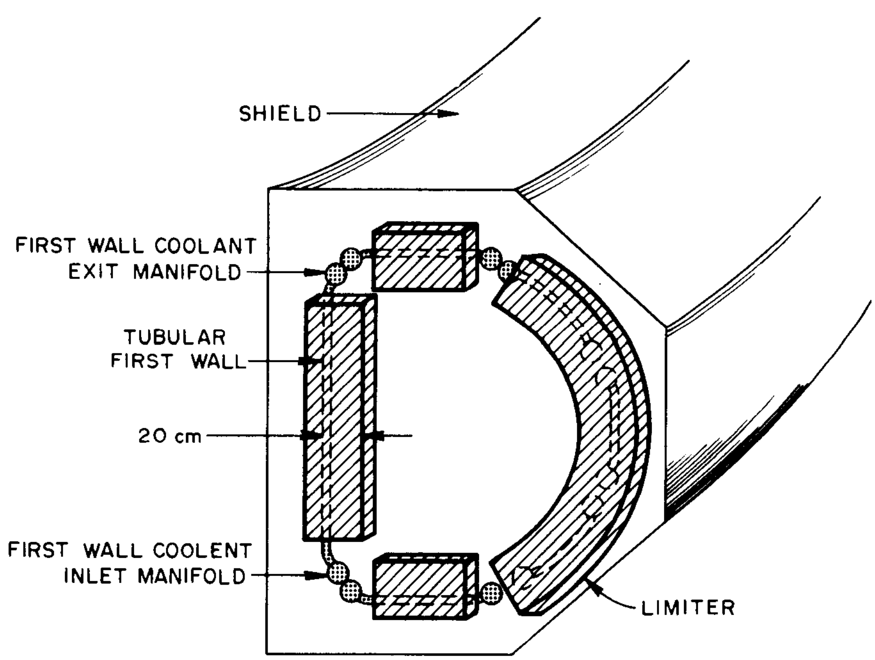

Fig. 3.1. Poloidal limiter.

A passively cooled limiter is assumed. An actively cooled system is avoided because of (1) the added complexity of supplying coolant to the limiter and (2) the possibility of coolant leaking to the plasma in the event of a limiter coolant tube burnout.

Graphite is the limiter material; its low-Z characteristics are desirable in regard to limiter contributions to plasma impurities. Graphite's thermal properties (high melting point, shock resistance, and low expansion) are all suitable for the operating environment. The graphite panels, which are 1-2 cm thick, are installed between adjacent first wall modules and extend 20 cm from the first wall toward the plasma. It is intended that they last for the life of the first wall. Their replacement would be part of the scheduled maintenance of the first wall. Limiters can be removed only by disassembling the torus.

4. FIRST WALL

4.1 FUNCTION

The primary function of the first wall, in conjunction with the divertor, is to absorb and remove the energy associated with the alpha particles, which makes up approximately 20% of the total plasma energy. The first wall is designed to take all the alpha particle energy. However, the actual heat load to the first wall is estimated to be equivalent to 50% of the alpha particle energy with the remaining 50% handled by the bundle divertor (see Sect. 10). Material damage associated with particle-surface interactions (e.g., sputtering, embrittlement, etc.) is taken by the first wall. The parameters of the first wall are listed in Table 4.1.

4.2 DESIGN DESCRIPTION

Type 316 stainless steel tubes, 2.5 cm in diameter, are brazed to type 316 stiffened sheet steel to make up modular first wall panels (see Fig. 4.1). Each of the faceted inner surfaces of the plasma chamber is lined with an independently cooled panel. The panels are mechanically

Table 4.1. First wall parameters

Construction	2.5-cm-diam stainless steel tubes
Maximum heat load (design value)	226 MW
Coolant	Water
Inlet temperature	40°C
Exit temperature	∿150°C
Actual heat load	113 MW
Neutron wall loading, L_W	2.38 MW/m^2
Maximum wall loading (surface heat)	0.60 MW/m^2
Actual wall loading (surface heat)	0.30 MW/m^2
Lifetime	4.5×10^5 cycles (10 years operation)

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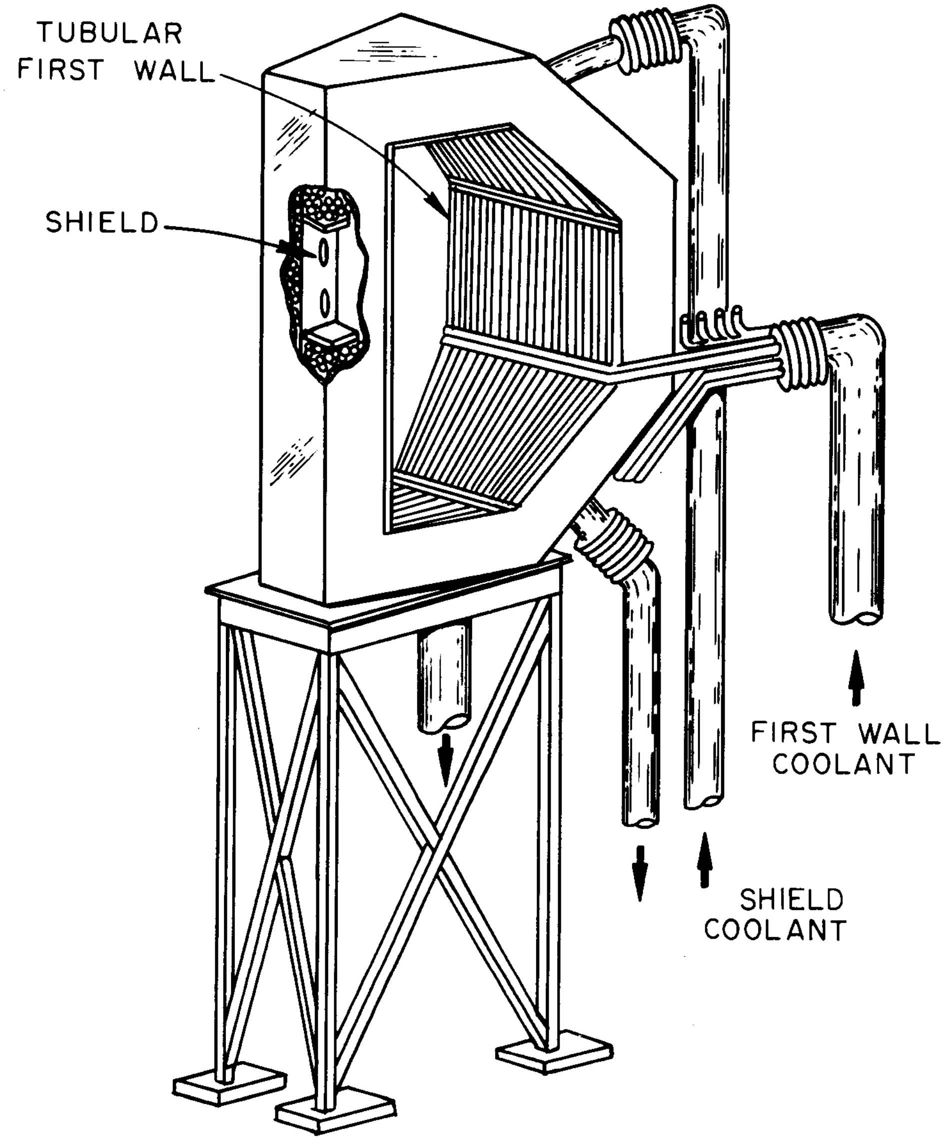

Fig. 4.1. First wall.

fastened to the chamber so that induced thermal stresses in the panels are minimal. Type 316 stainless steel was chosen because its physical properties are adequate to meet the design requirements and because its use presents an opportunity to expand the nuclear/thermal data base for this material.

The total first wall installation consists of 6 panels for each of the 16 sectors. Sectors may be removed from the device for replacement or repair (see Sect. 24).

The first wall is expected to last for the proposed 10-year life of the facility (based on the mission description in Ref. 1). Adherence to this schedule requires approximately 4.5×10^5 pulses. This number is within the low cycle fatigue limit of stainless steel even if the total alpha particle energy is deposited in the first wall. The actual heat load to the first wall is estimated to be 50% of the alpha particle energy (113 MW) with the remaining 50% handled by the bundle divertor. However, the coolant flow rate through the tubular first wall is adequate to transfer all the alpha particle energy (226 MW) at a temperature rise of approximately 110°C.

REFERENCE

1. D. Steiner, W. R. Becraft, T. G. Brown, W. A. Houlberg, A. T. Mense, Y-K. M. Peng, R. L. Reid, J. A. Rome, C. Sardella, T. E. Shannon, P. T. Spampinato, W. M. Wells, and G. W. Wiseman, *Oak Ridge TNS Program: Summary of FY 1978 Activities*, ORNL/TM-6720, Oak Ridge, Tennessee (to be published).

5. SHIELD

5.1 FUNCTION

The functions of the shield are to:

(1) reduce nuclear heating in the TF coils to acceptable levels,
(2) minimize activation of reactor components and auxiliary systems,
(3) attenuate radiation from activated components during maintenance shutdown periods,
(4) reduce radiation streaming around and through penetrations that view the plasma, and
(5) transfer the heat generated by attenuation of neutron and gamma fluxes by means of an active cooling system.

The shield parameters are listed in Table 5.1.

5.2 DESIGN REQUIREMENTS

The shield between the first wall and the TF coils will be thick enough so that the total time-averaged nuclear heating rate in the superconducting TF coils will be $\leqslant$20 kW (about one-third of the total pulse average heating rate in the TF coils).

Table 5.1. Shield parameters

Shield material	Stainless steel balls and borated water
Major radius, R	5.0 m
Shield minor radius	$\sim$1.5 m
Shield thickness	0.6 m
Shield thickness (around penetrations)	0.30 m
Number of sectors	16
Coolant	Low pressure borated water
Coolant heat load	1220 MW
Neutron wall loading, L_W	2.4 MW/m^2

5.3 DESIGN DESCRIPTION

The D-T plasma operation will produce large quantities of 14.1-MeV neutrons. These high energy particles, plus the secondary radiations they generate in the form of gamma rays and low energy neutrons, will cause damage to materials and represent a hazard to personnel. The shielding system of a D-T fusion device normally includes four components: (1) blanket, (2) bulk shield, (3) penetration shield, and (4) biological shield. The primary function of the blanket and bulk shield is to convert the kinetic energy of fusion neutrons and secondary gamma rays into heat. The additional function reserved for the blanket is breeding tritium; since TNS, at this time, does not include fuel generation, we are describing only a bulk shield having a dual purpose, energy conversion and reactor component protection by neutron attenuation. The penetration shielding surrounds the ducts for neutral beam injectors, divertors, vacuum pumps, fueling, and diagnostics. It provides virtually no attenuation for direct streaming but will protect components (such as the TF coils) that are adjacent to the penetrations. The biological shield is generally considered to be the concrete reactor cell (see Sect. 25), which provides the last barrier for neutron attenuation during fusion and limits activation levels outside the building to less than 100 mrem/year.

The bulk shield material is stainless steel in the form of 1-cm-diam balls with borated water acting as the coolant for each module. The minimum outer shield thickness (i.e., the shielding furthest from the bucking cylinder) will limit the PF coil organic insulation to a radiation level of 10^{10} rads over the life of the tokamak. The minimum inner shield thickness will limit the total time-averaged nuclear heating rate in the superconducting TF coils to an acceptable rate. A 5-cm thickness of lead is part of the bulk shield.

The total device shielding thickness was evaluated relative to the size and number of penetrations. The total effectiveness of the shielding in the reactor cell is greatly influenced by the area of the penetrations into the torus. The penetration shield around the neutral beam duct is 30 cm thick and the concrete thickness for the reactor cell is 2 m thick, assuming ordinary concrete.

6. MACHINE STRUCTURE

6.1 FUNCTION

The machine structure provides the structural support for the components that make up the tokamak device: the torus plasma chamber, the TF and PF coil supports, the dewar structure, and peripheral components that interface with the torus, such as the neutral beam injectors, the rf heating system, the pellet fueling device, and the divertor.

6.2 DESIGN REQUIREMENTS

The machine structure will be designed in compliance with accepted ASME safety codes to ensure a reasonable degree of conservatism and consistency in applying the design rules. Deviations will be permissible in areas where the codes do not apply and where revisions or extensions of the codes are warranted. The machine structure will be designed to complement the remote handling characteristics of the overall device, have a 10-year minimum design life, and incorporate materials that minimize neutron activation and have low magnetic permeability and high electrical resistance.

6.3 DESIGN DESCRIPTION

The major components that make up the machine structure are listed with corresponding design descriptions (refer to Fig. 6.1 to establish the location of each component within the tokamak device).

6.3.1 <u>Torus Plasma Chamber</u>

The torus plasma chamber surrounds the plasma, contains the bulk shield material, and provides support for the first wall. The first wall structure is mechanically supported off the inner wall of the sectors. The bulk shield material, consisting of stainless steel balls and borated water, is located in the interspace of the sector walls. The torus is made up of 16 sectors which are mechanically sealed and

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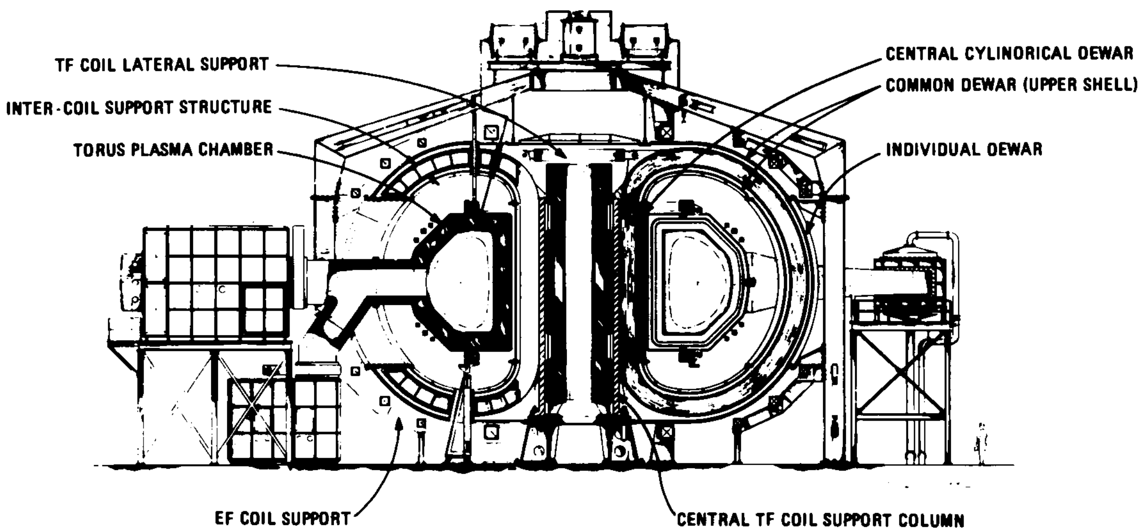

Fig. 6.1. Major components of the machine structure.

bolted at sector interfaces. Each 22.5° sector has a trapezoidal cross section and consists of three inner and outer, single-curvature-formed, stainless steel sheets separated by internal stiffeners and closed at each end. Four sectors form a quadrant (90°) of the torus and are supported from below by a curved beam attached to a truss structure connected to ground. The curved beam and truss structure are configured to allow a torus quadrant to be moved radially outward a small distance to provide initial intersector clearance during sector removal operations (see Sect. 24).

With all 16 sectors connected, a rigid structure is formed. Electromagnetically induced, radially outward forces are relatively small and are reacted by the intersector bolts. The interior D-shape coils of the poloidal field system (see Sect. 7.3), located at the top and bottom of the torus, act as bucking rings to support any electromagnetically induced centering forces. The torus itself is designed to withstand the 1-atm pressure differential required for shakedown operations.

6.3.2 Coil Support Structures

6.3.2.1 Central TF coil support column

The central support column is a cryogenically cold (5K) structure which (1) transfers the gravity forces of the TF coils and intercoil structure to a cryogenically isolated machine support base and (2) supports a portion of the magnetically induced centering and overturning forces of the TF coils based on the relative stiffness between the support column and the intercoil structure. The support column is a 12-sided polygon with a circular inner diameter. The structure has a vertical groove along the full height of each face to react a portion of the TF coil torsional forces. It is electrically segmented to reduce eddy currents.

6.3.3 Dewar Structure

The dewar structure forms a vacuum enclosure for the cryogenically cooled components that make up the tokamak device: the TF coils and intercoil structure, the OH solenoid, the interior EF coils, and the central support column. The dewars minimize tritium collection on the surfaces of these components and enable them to maintain their cryogenic condition when the vacuum building is repressurized, eliminating the dependence of the tokamak device on their cooldown and warmup cycles.

The dewar structure consists of two continuous stiffened shells, which enclose the upper and lower portions of the TF coils; a central cylindrical dewar, which encloses the inboard legs of the TF coils; and individual rectangular dewars, which enclose the outboard legs of each TF coil. The secondary vacuum environment provided by the vacuum building allows the use of mechanical connections to join the dewar sections.

Vacuum and thermal properties of the dewar system are discussed in detail in Sect. 9.

6.3.2.2 Intercoil support structure

The intercoil support structure, which reacts a portion of the magnetically induced TF coil forces, is a box beam construction located between coils and mechanically fastened to the TF coil structure along the upper and lower portions of the coils. The torsional load is balanced at the machine midplane by differential bending in the outer portion of the TF coil and by keying action at the bucking cylinder. A structural insulating shim of glass-reinforced epoxy acts as an interface between the TF coils and the intercoil structure to interrupt the eddy current path. Fault condition requirements for the failure of a single TF coil and the associated loads have not been defined; therefore, the extent of the intercoil structure required for this condition has not been addressed. The design shown indicates that out-of-plane fault loads generated along the unsupported section of the TF coil are beamed to the existing intercoil structure. If the stresses generated on the TF coil under this condition are unacceptable, it appears feasible to add local demountable supports to reduce the stresses without compromising the remote maintenance features of the device.

6.3.2.3 EF coil support

The intercoil forces associated with the EF coils are supported by an interconnecting structure between the EF coils. The upper support structure and machine floor support the gravity forces and any imbalance in the coil system.

6.3.2.4 TF coil lateral support

The TF coil lateral support member is a ring located at the top of the OH solenoid and connected by pins to the TF coils to support the imbalance in the TF coil gravity loads during initial assembly.

7. POLOIDAL FIELD SYSTEM

7.1 FUNCTION

The poloidal field (PF) system (1) induces a current in the plasma, (2) maintains the plasma columns in equilibrium both horizontally and vertically, and (3) protects the superconducting TF coils from magnetic flux changes. The parameters of the PF system are shown in Table 7.1.

7.2 DESIGN REQUIREMENTS

The PF system has a 10-year design lifetime for the main components and allows remote maintenance and replacement of the coils. Operating with rf preheating, the OH and EF coil systems will be capable of supplying 53 V-sec during startup, inducing a plasma current of up to 4 MA. Without rf preheating, 70 V-sec will be required. The OH system will also have sufficient volt-seconds capacity to provide a relatively long burn pulse, which would be adequate for viable reactor operation. The EF coils will supply sufficient vertical field to hold the plasma current in equilibrium during the burn phase.

7.3 DESIGN DESCRIPTION

An air core, hybrid PF system incorporating rf assisted startup was selected for the FY 1978 Reference Design. The PF system consists of a superconducting OH coil, interior (inside the TF coil bore) normal copper EF coils which carry 35% of the EF current, and exterior (outside the TF coil bore) superconducting EF coils which carry 65% of the EF current. The name "hybrid" is used to identify the mix of interior and exterior EF coils.

The size of the central bore for the Reference Design was taken from system studies which identified the geometric characteristics (i.e., the plasma size and major radius) for a minimum cost ignited tokamak reactor.[1] The OH system was sized for maximum utilization of

Table 7.1. PF system parameters

Primary OH coil	
General	
Life, t_L	10 years
Duty cycle, dc	96%
Inside radius of solenoid, R_I	0.875 m
Outside radius of solenoid, R_{os}	1.13 m
Length of central core, L_S	10 m
Number of turns (total), N_{oh}	2771
Mechanical	
Maximum hoop stress, σ_s	20,000 psi
Electromagnetic	
Ampere-turns	64 MAT
Maximum field at winding, B_{smax}	8 T
Field at coil axis, B_{sax}	8 T
Induced plasma current (by OH system), I_p	4 MA
Maximum rate of flux change in the OH coil, $\dot{B}$	1.3 T/sec
Coil swing time (full forward to zero current), t_s	6 sec
Coil volt-seconds (OH system)	59 V-sec
Stored energy, W	1.1 GJ
Coil voltage (maximum), V_c	26 kV
Maximum coil current/turn, I_c	23 kA
Conductor	NbTi
Current density overall, J_{OA}	2.5 kA/cm^2
Self-inductance (single turn), L_{oo}	0.37 μH
Thermal	
Conductor coolant	Helium
Total refrigeration required, P_{ref}	TBD[a]

[a]TBD = to be determined.

Table 7.1 (continued)

EF coil	
General	
Lifetime, t_L	10 years
Duty cycle, dc	94%
Number of coil systems, N_{sh}	6
Electromagnetic	
Ampere-turns	41 MAT
Coil volt-seconds	24 V-sec
Stored energy, W	2.6 GJ
Current/turn, I_{ac}	23 kA
Conductor material	
Normal conductor	Cu
Superconductor	NbTi

the central bore and for operation at a maximum field of 8 T. The PF system will supply 83 V-sec when operating with a half-biased OH system swinging from +8 T to -8 T, which is adequate for startup and 500 sec of burn.

Placement of the EF coils is important from the standpoint of maintaining plasma equilibrium and shape during the heating and burn phase, influencing power consumption, protecting the TF coils against plasma disruption, and servicing the coils. In the Reference Design, the EF coil system consists of six coil systems, named primarily for their locations with respect to the TF coil bore and their rough proximity to the plasma: interior inside coils (II), interior D-shape coils (ID), interior outside coils (IO), exterior inside coils (EI), exterior D-shape coils (ED), and exterior outside coils (EO). The PF system is shown in Fig. 7.1. As noted, the EF coils inside the TF coil bore are normal copper and those outside are superconducting. This arrangement of interior and exterior coils can maintain the plasma position and D-shape for a wide range of beta values.

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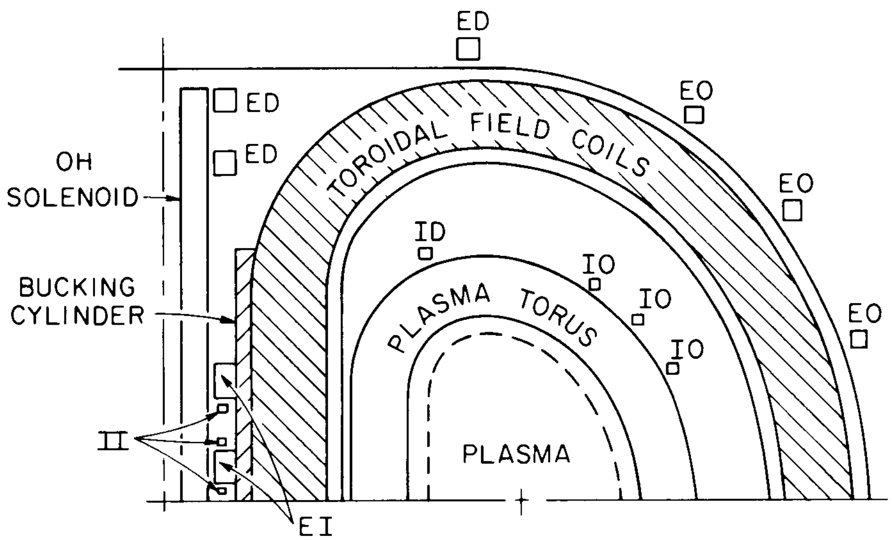

Fig. 7.1. PF coil locations.

Because of the intrinsic engineering difficulties with the assembly and repair of interior coils in the radioactive environment of a D-T tokamak reactor, an effort was made to reduce the number of turns of these coils. The 65%-35% split between exterior and interior EF coils offered a reasonable balance between coil maintenance, TF coil protection, and power requirements.

The time behaviors of the EF and OH coil currents are shown in Fig. 7.2, based on rf preheating during startup. The rf assisted startup was chosen because a substantial reduction is realized in the startup loop voltage and resistive volt-second losses, along with an increase in the startup time relative to operating without a rf preheating system. The rf preheating is supplied by five 200-kW gyroklystrons, which deliver 1 MW to the plasma during the 6-sec startup period.

REFERENCE

1. TNS Engineering Staff, *Four Ignition TNS Tokamak Reactor Systems — Design Study*, WFPS-TME-071, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania (October 1977).

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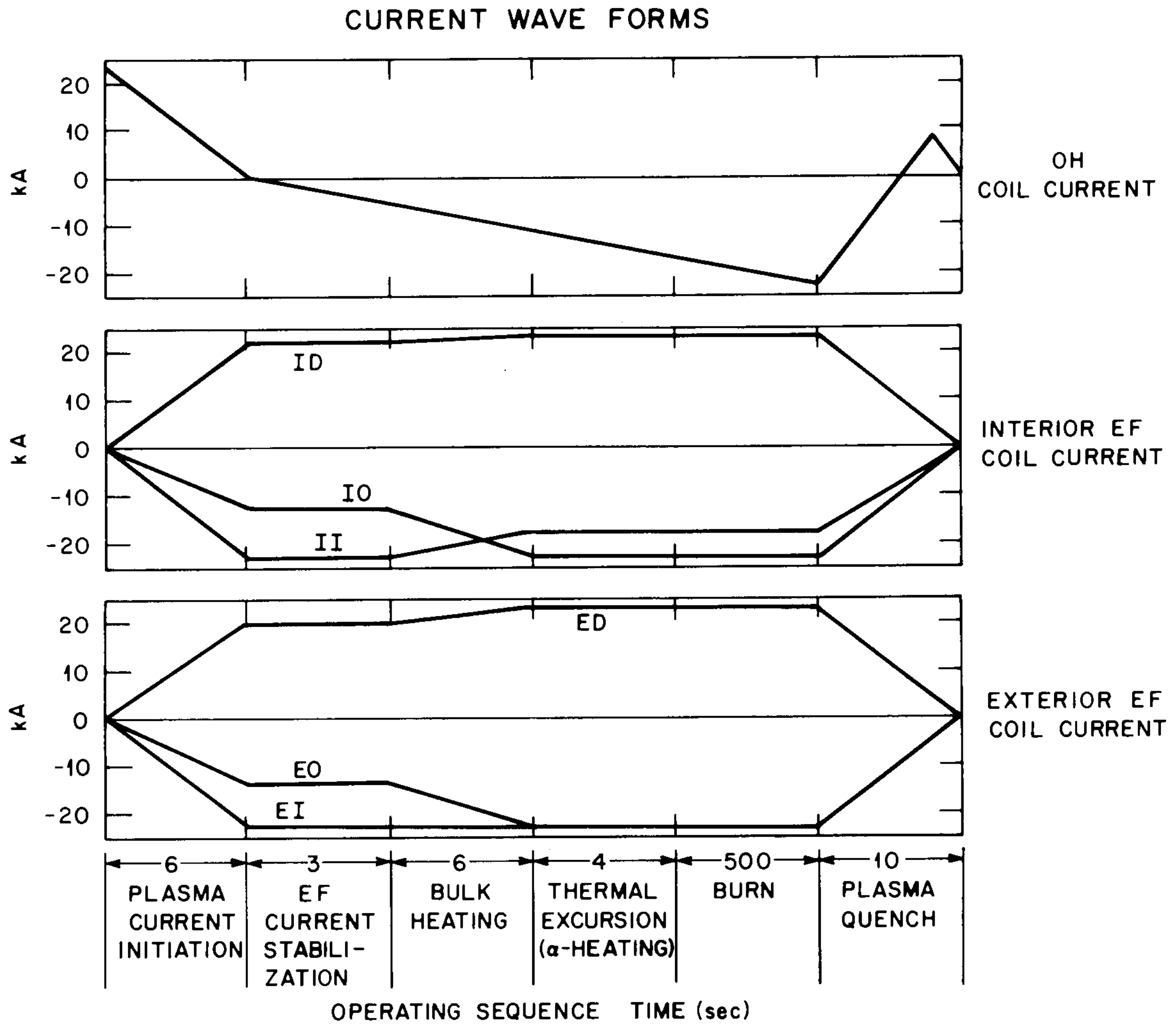

Fig. 7.2. PF coil current waveforms.

8. TOROIDAL FIELD SYSTEM

8.1 FUNCTION

The toroidal field (TF) system generates a toroidal magnetic field that combines with the pulsed poloidal magnetic field to confine the plasma. The system parameters are listed in Table 8.1.

8.2 DESIGN REQUIREMENTS

The TF coils must generate a magnetic field on axis (B_T) of 5.3 T and must have a magnetic field ripple at the plasma edge (δ_a) of <2.5% (peak to average). The duty cycle is 100%, and the coil lifetime is 10 years.

8.3 DESIGN DESCRIPTION

The required on-axis magnetic field of 5.3 T at a plasma major radius of 5.0 m is produced by 12 superconducting TF coils with a horizontal bore of 6.2 m and a vertical bore of 9.8 m. This combination of number of coils and coil size achieves a magnetic field ripple of 1.5% at the plasma edge, which is less than the maximum value of 2.5%. However, these larger TF coils, relative to the size required for 2.5% ripple, allow the PF coils inside the bore of the TF coils to be raised to parking orbits during torus segment removal and also allow space for the removal of a torus segment (consisting of one-sixteenth of the torus) between stationary TF coils.

The TF coils are a pure tension D-shape design with a trapezoidal cross section. The centering forces on the coils are reacted through a bucking cylinder and an intercoil support structure.

The superconducting elements are composed of an insulated, cabled conductor, using Nb_3Sn filaments in a copper stabilizing matrix. The cables are encapsulated by a thin stainless steel jacket and cooled by forced-flow supercritical helium, which runs through the interstices of

Table 8.1. TF coil system parameters

Geometric	
Number of coils, N_c	12
Coil shape	D
Horizontal opening, d_{tf}	6.2 m
Vertical opening, h_{tf}	9.8 m
Mean coil circumference, C	29.5 m
Plasma radius, a	1.2 m
Major radius, R	5.0 m
Electromagnetic	
Maximum field at winding, B_{max}	10.9 T
Field on axis, B_T	5.3 T
Total ampere-turns	132 MAT
Stored energy, W_c	18 GJ
Number of turns per coil	733
Structure-to-conductor area ratio	2.14
Current per turn, I_t	15 kA
Conductor	Nb_3Sn superconductor with copper matrix
Overall average conductor current density, JAC	37.5 MA/m^2
Structural	
Strain in conductor matrix, ε_m	<0.2%
Centering force per coil, F_c	470×10^6 N
Structural material	300 series stainless steel
Weight of each coil, W_{coil}	220,000 kg
Thermal	
Conductor coolant	Supercritical helium
Average helium inlet temperature, T_{in}	5.0K
Helium flow rate/coil, m	4 kg/sec
Total refrigeration required at 5.0K (includes pumping power), P_{ref}	48 kW

the strands. The conductors are pancake wound around a stainless steel coil support bobbin. There are eight slots in a bobbin with three Nb_3Sn superconductors in a slot. The TF coil configuration is shown in Fig. 8.1.

The superconducting TF coils are designed to be cryostable. The cryogenic recovery capability of the conductor design is defined as the ability to recover to a fully superconducting state from an initially imposed normal zone temperature (driven to that temperature by a sudden heat deposition in the conductor) over a given normal zone length. The present stability requirement ground rule specifies that superconductors shall be able to recover to their fully superconducting state after 10^5 J per cubic meter of conductor plus helium volume is suddenly deposited in half a turn of the conductor winding. Figure 8.2 indicates that the Nb_3Sn design can meet the requirement with a very small flow rate. However, in order to maintain a turbulent flow in the Nb_3Sn conductor channels, a flow mass velocity of 2 g/sec-cm^2 per conductor was selected. At this mass velocity, the stability margin of the Nb_3Sn superconductors is >3.

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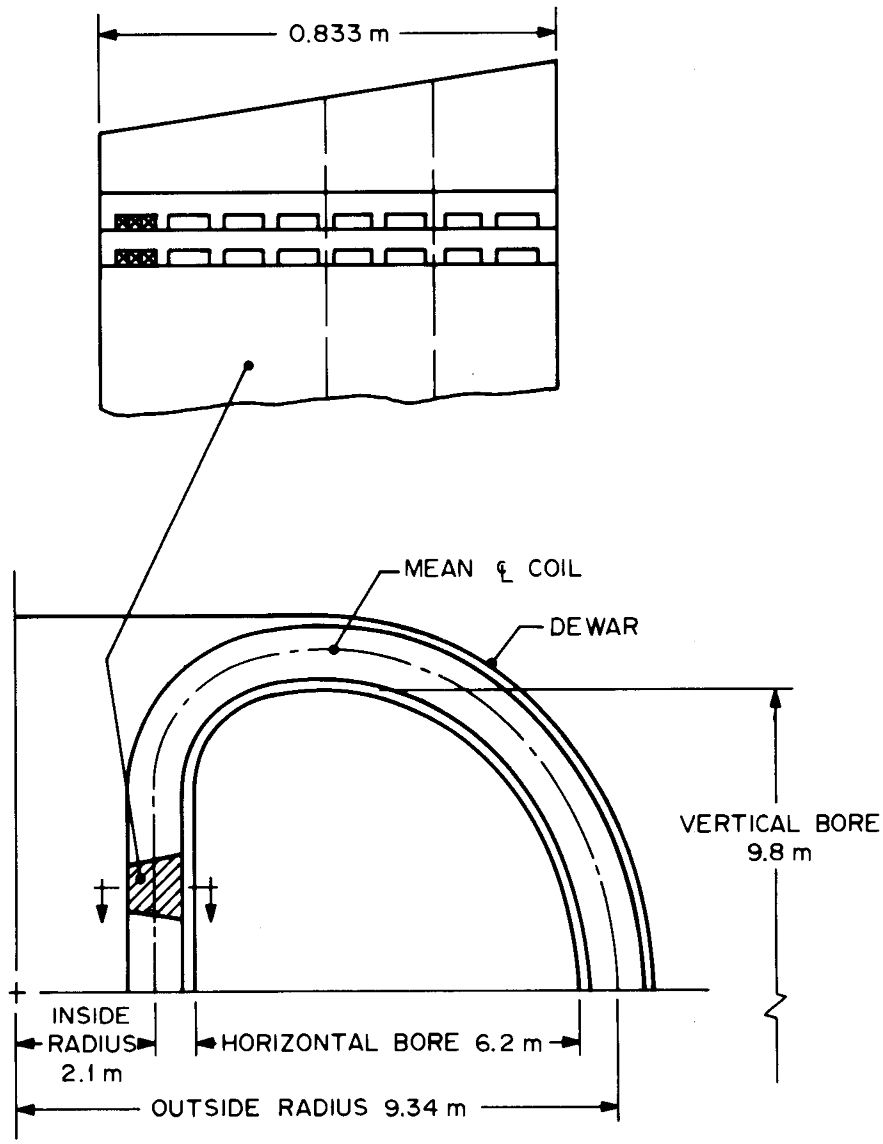

Fig. 8.1. TF coil configuration.

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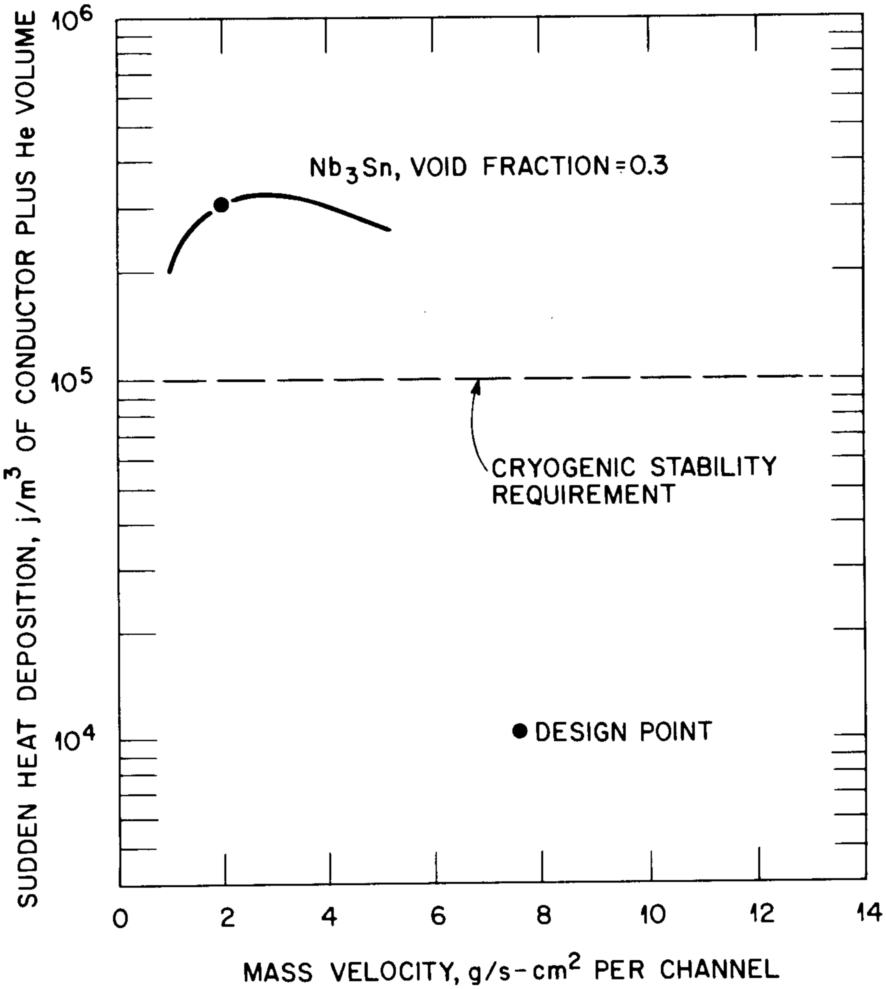

Fig. 8.2. Recovery capability of the TF coil superconductor.

9. DEWAR SYSTEM

9.1 FUNCTION

The dewar system reduces the heat leak into the superconducting coils and their supporting structures. It provides a vacuum barrier to reduce gas convection losses and a shield to reduce thermal radiation losses. The parameters of the dewar system are given in Table 9.1.

9.2 DESIGN REQUIREMENTS

The dewar system must enclose the TF coils, OH coils, central support column, and intercoil structure with a vacuum-tight enclosure. Vacuum leak rates of the dewar system must be sufficiently low to allow 6 months of operation (the assumed time between normal maintenance periods) in an evacuated reactor cell without surface conditioning to remove particle accumulation. Vacuum pumping will be used for initial pumpout and for collection of desorbed gases during surface conditioning.

Under normal operating conditions, the dewar system will be supported from ambient temperature surfaces only, to minimize solid conduction heat losses. The dewar system will be designed to withstand external pressure of 1 atm. Contact between the dewar structure and cold surfaces via standoffs is permitted under external pressure loading.

9.3 DESIGN DESCRIPTION

The dewar is a vacuum-tight structure which encloses the TF coils, the central support column, and the OH coils inside the toroidal bore. Three geometric configurations are used for the dewar system components. A cylinder encloses the inboard legs of the TF coils, rectangular sections enclose the outboard legs, and shell structures enclose the upper and lower portions of the TF coils. The dewar sections are joined by four rings attached to the upper and outer support structure. Figure 9.1 shows these components in relation to the device. Figure 9.2 shows an individual dewar.

Table 9.1. Dewar system parameters

Internal pressure	10^{-5} torr (maximum)
External pressure (normal operation)	10^{-4} torr
Thermal shield	
Temperature	80K
Coolant	LN_2
Surface area (one side)	1450 m^2
Coolant flow rate	TBD[a]
Insulation	
Location	Off warm surface
Type	Superinsulation
Thickness	TBD
Heat leak	TBD

[a]TBD = to be determined.

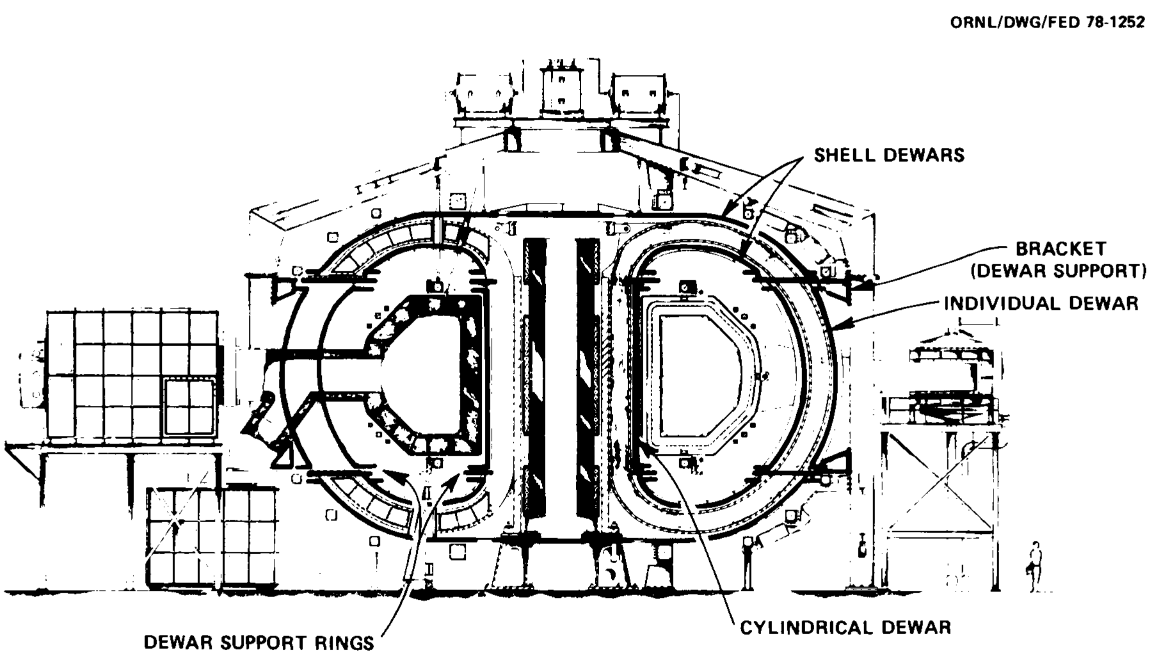

Fig. 9.1. Dewar locations.

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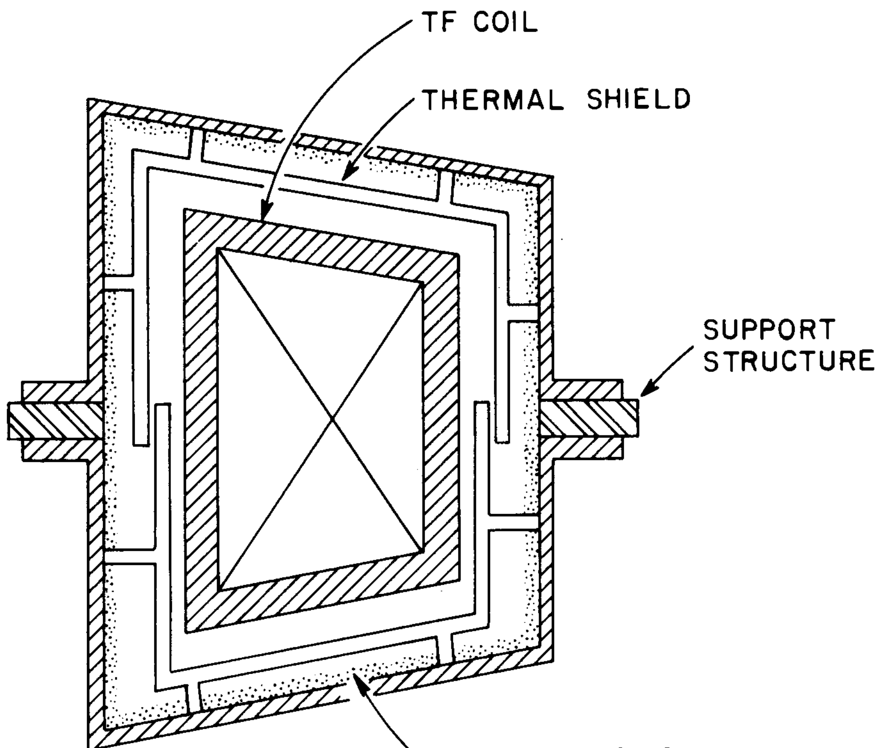

Fig. 9.2. Individual dewar.

Under normal operating conditions, there is no pressure differential across the dewar; however, during maintenance operations, when the reactor cell is at atmospheric pressure, relatively high pressure loading will result. These loads are reacted by standoffs between the inner and outer shell dewars and by standoffs to the coil surfaces for the cylindrical and individual dewars. Dewar sections will be joined primarily by welding; however, bolted construction will be used where maintainability is enhanced by it.

Because the dewar operates in an evacuated cell, absolute vacuum-tightness is not required. The permissible dewar leak rate is determined by the maximum particle buildup that the cryogenic surfaces can tolerate without degradation of thermal properties and by the requirement for 6 months of continuous operation between maintenance periods. During maintenance shutdown, dewars will be purged by warming up the coils to

80K and pumping out desorbed gases. Leakage into the dewar will be higher by several orders of magnitude when the reactor cell is at atmospheric pressure, and intermittent operation of the pumps will be required to maintain a base pressure of approximately 10^{-2} torr as dictated by refrigeration requirements at 80K.

10. DIVERTOR SYSTEM

10.1 FUNCTION

The divertor system (1) minimizes the interaction between plasma particles and the walls of the plasma containment vessel and (2) diverts and collects impurities emanating from the walls. As a consequence, it is necessary to collect the diverted particles and to handle the heat load associated with these diverted particles. For design purposes, it has been assumed that the upper limit on the heat load to the divertor is equivalent to the alpha particle energy. The actual heat load to the divertor is estimated to be 50% of the alpha particle energy, with the remaining 50% absorbed by the first wall. The divertor system parameters are listed in Table 10.1.

10.2 DESIGN DESCRIPTION

The bundle divertor shown schematically in Fig. 10.1 is the type chosen for the Reference Design. It is composed of two coils, side by side, with opposite current flows that provide a strong short-range magnetic field, which is superimposed on the toroidal field in the plasma to produce zero field close to the external surface of the plasma. The resulting field distortion allows a bundle of field lines to be extracted and form a loop external to the plasma along which charged particles trapped on the lines can travel.

In a reactor environment, the divertor will be subjected to a high level of radiation, and some shielding (on the order of 1 m for superconducting coils) will be required to protect conductors and insulation from cumulative damage. This shielding requirement leads to oversize divertor coils, which create excessive perturbation of the field lines in the plasma. However, divertor coils of the proper size (i.e., the size which will not produce excessive perturbation of the plasma field lines) cannot be adequately shielded. A shield thickness of 0.4 m appears to be a feasible compromise. This thickness precludes the use of superconducting and normal cryogenic coils but permits the use of water-cooled copper with a shortened life.

Table 10.1. Divertor system parameters

Geometric	
Number of bundle divertors	1 operating, 1 or 2 on standby or being cleared of deposited lithium
Divertor opening	0.4 m
Shield thickness	0.4 m
Flux bundle expansion ratio	90
Performance	
Ripple at plasma centerline (due to divertor)	3%
Plasma edge particle loss rate (energy transport)	3×10^{23} particles/sec
Plasma fraction recycled at edge as neutrals	85%
Maximum heat load (design value)	226 MW
Actual heat load	113 MW
Electromagnetic	
Coil type	Water-cooled copper
Ampere-turns	12×10^6
Thermal	
Heat flux to divertor collector	10 MW/m^2
Divertor collector surface area	22.6 m^2
Lithium temperature rise	114°C
Divertor coil coolant	Water
Divertor coil ohmic heating	70 MW
Shield coolant	Water
Particles pumped by divertor	4.6×10^{22} particles/sec

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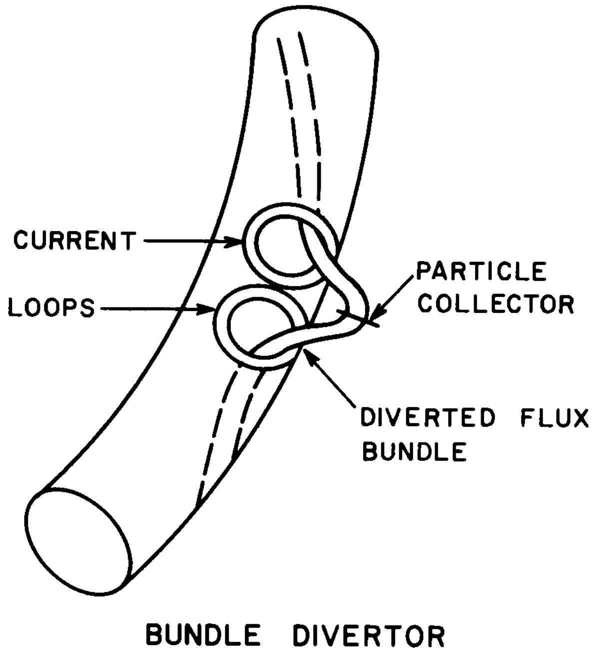

Fig. 10.1. Bundle divertor.

The divertor will expand the diverted magnetic field to reduce particle and energy flux. For the purpose of design, it is assumed that the divertor must absorb all the power associated with the alpha particles (226 MW). This power is assumed to leave the plasma through a particle loss rate at the plasma edge of 3×10^{23} particles/sec. It is further assumed that 85% of these particles recycle as neutral gas. The number of plasma particles that the divertor is required to pump is therefore 4.6×10^{22} particles/sec.

For a solid absorber, the getter material must either constitute or be a part of a heat exchanger, which requires material having good strength and thermal conductivity, since it will be subjected to at least 10^5 cycles of absorption and desorption. Consideration of data on hydrogen embrittlement of structural materials (for one absorption cycle) suggests that this will seriously degrade the material. Also, the times and temperatures for regeneration of getters (700-1000°C for several hours) tend to indicate infeasibility. Therefore, these functions will be accomplished by an array of lithium jets, shown schematically in Fig. 10.2. For the absorption of a heat flux of 10 MW/m^2, there must be 22.6 m^2 on which the expanded, diverted magnetic field impinges. The expansion requirement leads to a system that has much in common with the direct conversion approaches proposed for advanced fuel concepts.

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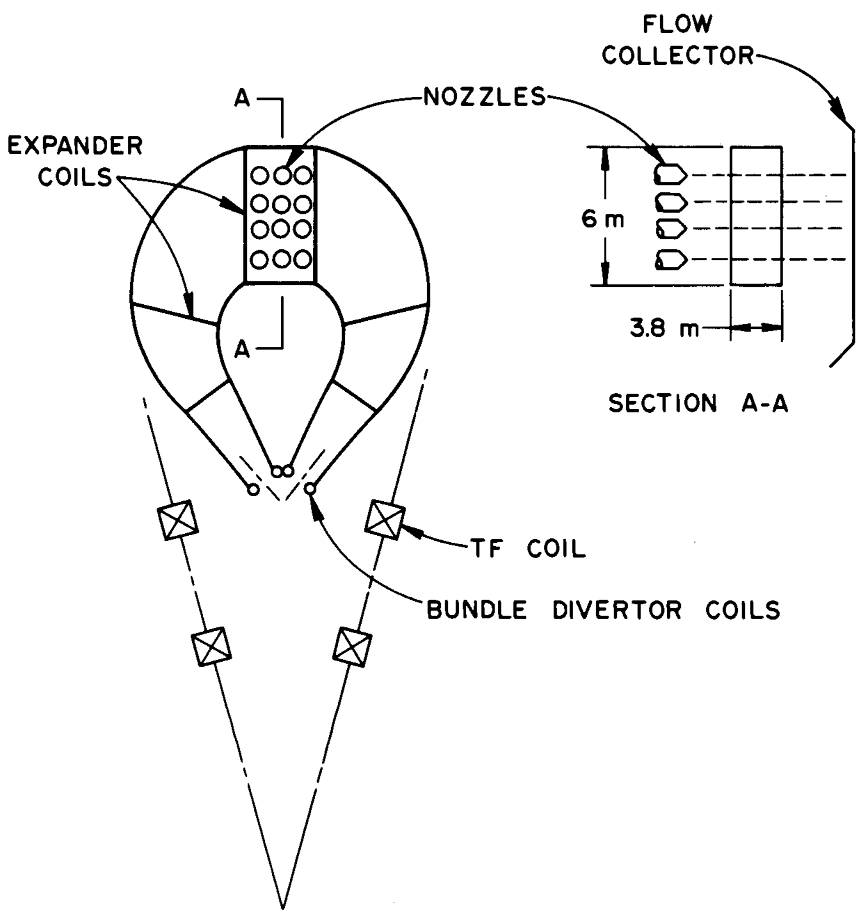

Fig. 10.2. Bundle divertor with lithium jet energy collection system.

11. BULK HEATING SYSTEM

11.1 FUNCTION

The bulk heating system raises the tokamak plasma temperature from that achieved by rf assisted ohmic heating to the ignition level. For this purpose, neutral beam injection has been chosen. The system parameters are listed in Table 11.1.

11.2 DESIGN REQUIREMENTS

The design requirements for the bulk heating system call for a total injected power of 50 MW with a beam energy of 150 keV. The pulse length is ≤ 6 sec and the duty cycle is 1%. The system is designed to last for the 10-year lifetime of the machine.

11.3 DESIGN DESCRIPTION

Because OH techniques alone are not adequate to provide ignition temperatures in plasmas, they must be assisted by neutral beams and/or by rf heating. The Reference Design uses four beam lines, with three sources per line, injecting through four rectangular apertures into the plasma. Each beam line includes a positive ion source, a neutralizer region, a direct energy recovery system, and a drift region to connect the beam line to the torus. The total electrical efficiency of the beam line is estimated to be 51%.

Previous considerations pointed to the expected need for high energy beams to penetrate to near the plasma center. Present calculations seem to support the systems advantages of using injection scenarios that call for moderate energy (120- to 150-keV) beams. This avoids the use of high power neutral beams with particle energies greater than 300 keV, for which the technology is still evolving. The efficiency of such high energy positive ion source beam lines becomes quite low and requires large quantities of power. Negative ion sources, which have the potential for larger efficiencies, are in the early stages of development and are five years or more behind the positive ion sources in possible applications.

Table 11.1. Bulk heating system parameters

Structural/configurational	
Sources per beam line	3
Total sources	12
Number of beam lines	4
Size of beam apertures into torus	0.4 m × 1.2 m
Injection angle (perpendicular to plasma surface)	⩾16° (in the direction of plasma current)
Performance	
Source current	100 A
Full energy power/source	3.9 MW
Half energy power/source	0.4 MW
Total power/source	4.3 MW
Total full energy injected power capability	46.8 MW
Total injected power capability	51.6 MW
Total power for particle acceleration	101 MW
Beam energy	150 keV
Full/half energy	0.90/0.10
Neutralization fractions (full energy/half energy)	0.33/0.70
Geometric transmission	0.88
Pulse length	⩽6 sec
System power efficiency	51%

The systems needs growing out of these data have provided the impetus to evolve startup scenarios, such as small radius buildup and plasma density buildup, which avoid the need for high energy beams. For the Reference Design plasma, a full bore, low density startup is feasible. This scenario requires injecting 50 MW of 150-keV beams into the plasma for approximately 6 sec.

Direct recovery of the energy in the unneutralized particles in a beam line is being developed and is considered a part of the Reference Design. This technique should allow the achievement of efficiencies higher than 51% in the 150-keV positive ion source beam lines.

12. RF HEATING SYSTEM

12.1 FUNCTION

The rf heating system supplies high power microwave energy to the plasma to initiate ionization and heating during startup. The rf assisted startup is expected to significantly reduce OH power requirements and cost. System parameters are shown in Table 12.1.

Table 12.1. RF heating system parameters

Structural/configurational	
Number of gyroklystron tubes	5
Tube length	1.22 m
Tube diameter	0.91 m
Performance	
Total injected power	1 MW
Pulse length	2 sec
Frequency	120 GHz
Injected power per tube	200 kW
Efficiency (gyroklystron device)	30%
Total electrical demand (from grid)	7.5 MVA

12.2 DESIGN REQUIREMENTS

The design requirements for the rf heating system call for 1.0 MW of injected power delivered at a frequency of 120 GHz. The pulse length is 2 sec and the duty cycle is 0.4%. The system's lifetime is 10 years.

12.3 DESIGN DESCRIPTION

The use of microwave heating in the Reference Design is considered to be a promising approach to preionization and may significantly lower the power requirements and cost of the OH power supply systems. Preionization by rf heating can substantially reduce the OH power supply energy needed for large loop voltage breakdown. Power transmission is accomplished by relatively efficient waveguides. The proposed system is based on a configuration of five 200-kW gyroklystrons which will deliver 1 MW to the plasma for approximately 2.0 sec.

Five power amplifiers driven by a single master gyrotron oscillator generate the rf power. The gyrotron is a new type of microwave vacuum tube based on the interaction between an electron beam and microwave fields where coupling is achieved by the cyclotron resonance condition. This type of coupling allows the beam and microwave circuit dimensions to be large compared to a wavelength. Power density problems encountered in conventional traveling wave tubes and klystrons at millimeter wavelengths are not as severe in the gyrotron. Each gyroklystron amplifier tube has a minimum gain of 30 dB and produces 200 kW. The single master oscillator operates at the fundamental frequency, produces 1 to 2 kW, and drives the individual power amplifiers through a 5:1 divider network. Each power amplifier tube is 1.22 m long by 0.92 m in diameter, requires several electrical power supplies and oil, water, and fluorocarbon coolants and has a superconducting focusing magnet. The rf power is transmitted through oversize waveguides (i.e., the waveguide diameter is approximately ten times the wavelength) to penetrations in the torus.

Five shielded canisters, each containing a high power rf source, are to be mounted on a support platform on top of the tokamak.

13. ELECTRICAL ENERGY STORAGE SYSTEM

13.1 FUNCTION

The electrical energy storage system supplies the total electrical energy demands during the pulsed operation of the device and buffers the utility from peak power demand required during operational pulsed scenarios. The system parameters are listed in Table 13.1.

Table 13.1. Electrical energy storage system parameters

Vertical shaft MGF sets	
Total MVA	600 MVA
Lagging power factor	0.7
Energy	4.3 GJ
Frequency at maximum speed	90 Hz
Generator voltage	15 kV
Full load	13.8 kV
Total full load pulses	10^6

The peak power demand occurs at different times in the OH and EF systems. The OH circuits, taken as one group, and the six EF circuits, taken as another group, are each sized for the peak 600-MVA demand. The poloidal systems, using superconducting coils and an optimized electrical design, operate off the utility line during the burn phase. During the startup and shutdown phases, the peak power demand is supplied by two motor-generator flywheel (MGF) sets. The MGF sets are vertical shaft sets with a combined rated capacity of 600 MVA at a lagging power factor of 0.7 and a combined deliverable energy of 4.3 GJ over a 35-sec interval every 560 sec.

The TF coils are superconducting coils that are precharged to maximum current (15 kA) off the utility line over a period of time (30 min) sufficient to minimize power supply size and are then maintained at peak current continuously. As a result, no energy storage is required for the TF coils.

13.2 DESIGN REQUIREMENTS AND DESCRIPTION

The generators will be started using standard wound rotor induction motors and then driven up to rated maximum speed (corresponding to some frequency equal to 90 Hz on the generator) with a cycloconverter system in conjunction with the rotor induction motor directly coupled to the generator. The MGF sets are capable of providing the total pulsed load demanded by the PF coils during startup and heating. The electrical load demanded by the PF coils during the 500-sec burn is taken directly from the utility grid. The MGF sets are also capable of accepting the stored energy from the PF coils.

The MGF exciter capacity is sufficient to control the generator voltage during a load change from no load to rated pulse capability. A combination of regenerative and dynamic braking is required to stop each MGF system within approximately 30 min. In addition, mechanical braking is required to bring each unit to a complete stop from 10% of its rated speed. Parallel operation of the two MGF sets and operation at reduced capacity with one MGF set will be possible. Remote control and monitoring are necessary.

The MGF units are vertically mounted units installed in concrete pits, with the flywheel as an integral part of the generators. The design lifetime of the MGF units is on the order of 25 years, which translates into approximately 1 million pulses.

14. ELECTRICAL ENERGY CONVERSION SYSTEMS

14.1 FUNCTION

The electrical energy conversion systems convert ac power to dc power for use by the TF, OH, and EF coil systems. The ac power from the MGF units (see Sect. 13) is transferred via the primary energy distribution system (see Sect. 15) to the conversion systems. There are three conversion systems, one for each set of coils:

(1) the toroidal field energy conversion (TFEC) system,
(2) the ohmic heating pulsed energy conversion (OHPEC) system, and
(3) the equilibrium field pulsed energy conversion (EFPEC) system.

The system parameters are listed in Table 14.1.

14.2 DESIGN REQUIREMENTS

The TFEC system must provide sufficient no-load dc voltage and current to the TF coils to achieve a toroidal field of 5.3 T at the major radius (5 m). The system must be able to maintain this current continuously.

The OHPEC system must provide a no-load dc voltage of 26 kV and a peak full load current of 23 kA to the OH coils to induce a plasma current of 5 MA.

The EFPEC system consists of six separate circuits to serve the six EF coil systems (see Sect. 7.3). The required voltage and current for each system are listed in Table 14.1.

Each energy conversion system will be operated by a digital computer for all desired scenarios, with local microprocessor control of shutdown, maintenance, and system status. During the plasma current takedown portion of the normal operating cycle, the OHPEC and EFPEC systems will reduce the coil currents by applying a negative voltage and will regeneratively return most of the stored energy to the MGF sets (see Sect. 13).

Table 14.1. Electrical energy conversion systems parameters

TFEC system	
No-load dc voltage	2 kV
Peak current	15 kA
Field strength (at R = 5 m)	5.3 T
Current regulation at full load (% rated current)	0.5%
Range of controlled operation	
Minimum (% rated current)	10%
Maximum (% rated current)	100%
OHPEC system	
No-load dc voltage	26 kV
Peak current	23 kA
Total volt-seconds	29 V-sec
Plasma current	5 MA
Current regulation at full load (% rated current)	0.5%
Range of controlled operation	
Minimum (% rated current)	10%
Maximum (% rated current)	100%
EFPEC system	
II coil system	
No-load dc voltage	2 kV
Peak current	23 kA
ID coil system	
No-load dc voltage	2 kV
Peak current	23 kA
IO coil system	
No-load dc voltage	4 kV
Peak current	23 kA
EI coil system	
No-load dc voltage	6 kV
Peak current	23 kA

Table 14.1 (continued)

EFPEC system	
ED coil system	
No-load dc voltage	6 kV
Peak current	23 kA
EO coil system	
No-load dc voltage	10 kV
Peak current	23 kA
Current regulation at full load (% rated current)	0.5%
Range of controlled operation	
Minimum (% rated current)	10%
Maximum (% rated current)	100%

14.3 DESIGN DESCRIPTION

The basic power supply element in all three energy conversion systems is the power convertor system. This system precharges the TF coils off the utility line and supplies the PF coils with a predetermined current pulse lasting up to 535 sec and repeated with full power every 560 sec.

Each energy conversion system consists of a number of power convertors serially interleaved with a corresponding equal number of TF, OH, or EF coil sets to minimize fault potentials and insulation ratings. Figure 14.1 is a simplified block diagram showing the electrical path from the utility to a typical PF coil. The fixed and variable frequency buses feed a grid of power convertors and transformers necessary for the OHPEC and EFPEC systems. (For simplicity, the coil/power convertor interleaving and the many transformers are not shown.)

14.3.1 Power Convertor System Components

The power convertors can operate in either rectifier or inverter modes and are constructed in modular fashion for maximum flexibility and

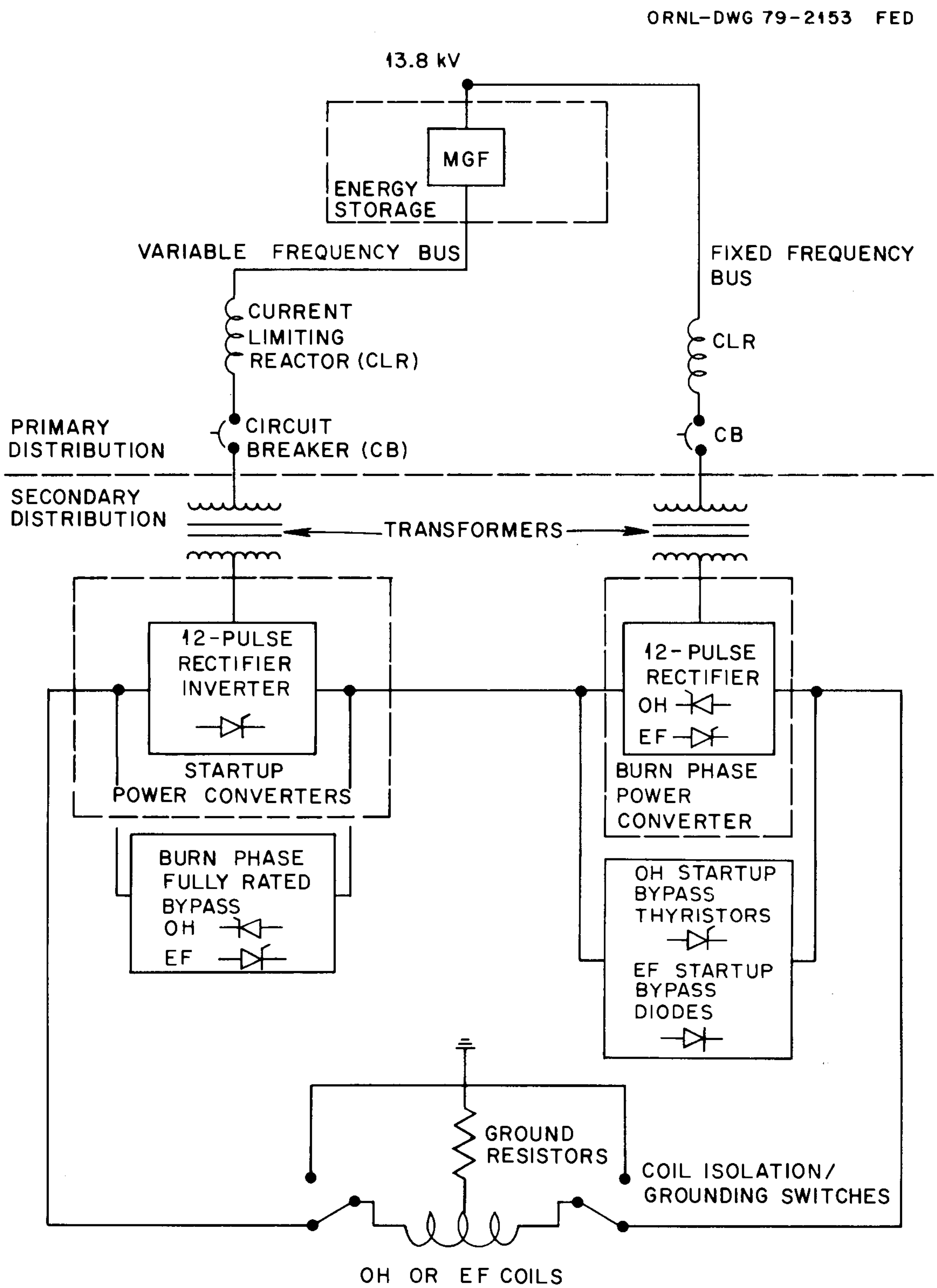

Fig. 14.1. Electrical path of the energy storage and conversion systems.

interchangeability between systems. Each power convertor consists of a 2-kV power supply cabinet, a power supply controller cabinet, and auxiliary equipment cabinets.

14.3.1.1 Power supply

The power supply is equipped with firing circuits and is controlled from its associated power supply controller. The power supply consists of two sections connected in series and configured as a 12-pulse rectifier-inverter supplied by a 3-winding transformer. The two sections may be connected in parallel by rearranging the dc buswork. Each section is a 6-pulse rectifier-inverter rated at 1 kV and the total coil current. Each section contains a sufficient number of power modules (common to all systems) connected in parallel to attain the proper current; the parallel groupings are connected in series to attain the proper voltages for the PF coil systems. The section is also equipped with separate fully rated bypass modules.

The power module is a 3-phase thyristor bridge with enough water-cooled thyristors to handle 3-5 kA in a tray arrangement. The power module is the basic element common to all the power supplies in the electrical energy conversion system and allows interchangeability among the TFEC, OHPEC, and EFPEC power convertors.

The fully rated bypass modules commutate the full load current out of the power modules for all serious convertor faults. During the burn phase, capability exists to provide a serial current bypass and allow a small burn phase power supply to drive each poloidal coil system. During startup and preburn plasma stages, the serial current bypasses the burn phase power supplies via bypass diode stacks in each EF circuit (see Fig. 14.1).

The OH circuit (also shown in Fig. 14.1) requires a bypass thyristor stack (at the low burn phase voltage) to control the bypass of the OHPEC system during startup. During burn, the current is in a direction opposite to that in startup, and the OH burn phase fully rated thyristor stacks (in series with the burn phase power convertor in the same current direction) provide the serial current bypass around the large OH startup convertor system. The OH startup and burn phase bypasses are thyristors in order to handle opposite current flows.

14.3.1.2 Power supply controller

Each power supply controller must provide control of the following functions:

(1) SCR firing reference voltage indicator (from power supply transformer primary),
(2) ac supply system status,
(3) ac supply disconnect switch signals,
(4) ac supply circuit breaker trip signal,
(5) power module bridge currents (delta/wye),
(6) power supply cooling water flow,
(7) power supply section output voltage,
(8) power supply interlock,
(9) normal/bypass power module SCR gate signals,
(10) power supply status,
(11) ground fault relay signals,
(12) crowbar control signal, and
(13) ground switch control signal.

14.3.1.3 Auxiliary equipment

The auxiliary equipment for the power convertor system consists of the following:

(1) manually operated selector switches to aid in maintenance and to bypass failed units,
(2) motor operated dc grounding switch to automatically ground the dc buses in emergency or personnel protection situations,
(3) dc surge divertor to limit line-to-line and line-to-ground switching type surges and transient overvoltages,
(4) dc transient suppressor to limit the dv/dt caused by system disturbances,
(5) ground fault protection system,

(6) voltage measurement,

(7) current measurement, and

(8) mechanical crowbar.

14.3.2 Power Convertor System Considerations

The large, quickly changing current variations during the plasma startup phase and the heating effects of coil current pulse scenarios dictate the size of the power convertor systems. However, there is considerable power dissipation in the power convertor systems themselves, in addition to that resulting from interconnecting buswork and contacts during the burn phase. For superconducting coil systems, the predominant areas of power dissipation during the burn phase are the power convertor system and the busbar. A separate, small burn power supply close to the coils can be designed to drive reduced parasitic losses and results in a significant reduction in power dissipation with low utility power drain, allowing direct operation off the utility line during burn. These considerations are integral to extended burn devices and reactor economics.

15. PRIMARY AND SECONDARY ELECTRICAL DISTRIBUTION SYSTEMS

15.1 FUNCTION

The primary electrical distribution system receives ac power from the utility (supplied from several 138-kV substations) and transforms 600-MVA ac power to supply a 13.8-kV load bus. The secondary electrical distribution system receives dc power from the electrical energy conversion systems (see Sect. 14) at appropriate voltages and currents to provide power to the TF and PF coils. The system parameters are listed in Table 15.1.

Table 15.1. Primary and secondary electrical distribution system parameters

Utility supply	
Substation voltage	138 kV
Demand	600 MVA
Primary load bus voltage	13.8 kV

15.2 DESIGN REQUIREMENTS AND DESCRIPTION

The primary and secondary electrical distribution systems will be separated into a number of separate circuits or buses so that rated electrical capability is reliably maximized in the event of an isolated equipment failure. The primary electrical distribution system will have current limiters to limit short-circuit current and voltage limiters to isolate equipment failures. The secondary electrical distribution system will have interchangeable interconnection elements, which provide flexibility by allowing bypass of faulted equipment and operation at reduced levels.

16. EMERGENCY STANDBY POWER SYSTEM

16.1 FUNCTION

The emergency standby power system is available in the event of failure or interruption of the utility service, the local low power circuits, or the variable frequency load bus (supplied by motor-generator flywheel sets and used to provide peak ac power to electrical energy conversion equipment). The emergency standby power system provides dc power to all energy conversion systems to allow a safe shutdown with full removal of all coil energy.

16.2 DESIGN REQUIREMENTS

The emergency standby power system will be independent of the utility and of the energy storage system (see Sect. 13). It will have its own uninterruptible energy storage system, consisting of batteries at ground level or insulated from ground at some higher potential (as required by the power convertors' firing circuits). Each power supply controller and each set of thyristor firing circuits will have its own hard-wired battery storage system with its own trickle charger.

The emergency standby power system will receive ac power from the utility and provide sufficient energy to terminate any operating scenario or permit completion of the plasma discharge, whichever is determined to be necessary.

16.3 DESIGN DESCRIPTION

In the event of failure of the utility or the main energy storage units, or any interruption of service, the emergency standby power system will provide sufficient energy (approximately 25 MJ) from its energy storage system to controllably terminate any operating scenario or permit completion of the discharge. The safe shutdown of an operating scenario includes discharging all energy stored in the system without causing the ac breaker to trip, fuses to blow, or any repetitive ratings to be exceeded.

17. TORUS VACUUM PUMPING SYSTEM

17.1 FUNCTION

The torus vacuum pumping system (TVPS) evacuates the plasma chamber from atmospheric pressure to a base operating pressure of 10^{-6} torr and maintains base pressure by removing spent gases after each burn. The system parameters are listed in Table 17.1.

Table 17.1. Torus vacuum pumping system parameters

Torus volume	340 m^3
Torus inside area (smooth)	380 m^2
Outgassing rate	4×10^{-3} torr-liters/sec
Base pressure (prefill)	1×10^{-6} torr
Base pressure (postfill)	1×10^{-4} torr
Pumpout time (postpulse)	25 sec
High vacuum system	
Pump type	Cryosorption
Number of pumps	4
Effective speed	2.5×10^4 liters/sec
Specific speed per pump	2.0×10^5 liters/sec
Cryopanel area	2.0 m^2
Duct size	120 cm × 80 cm × 700 cm
Roughing system	
Pump type	Oil-free mechanical
Number of pumps	4
Pump speed (per pump)	75 liters/sec
Foreline pressure	10^{-2} torr
Gas collection system	
Roughing pump discharge	Pressurized tank
High vacuum pump discharge	Zr-Al getters

17.2 DESIGN REQUIREMENTS

The TVPS is composed of four independent pumping units, each consisting of a high vacuum pump, a roughing pump, and a gas collection system (see Fig. 17.1). The high vacuum pumps will maintain a base operating pressure of 10^{-6} torr against the outgassing load of the torus interior surfaces. They will also remove spent gases and restore base pressure within 25 sec after each burn. The TVPS must operate as a closed system; all pumped gases will be collected and retained for further processing (see Sect. 20). Roughing pumps, transfer pumps, and other mechanical equipment will be located in the basement or some other area having an atmospheric environment.

17.3 DESIGN DESCRIPTION

The high vacuum pumps are cryosorption units. Each of the four cryosorption pumps is tapped into a neutral beam injector duct to minimize the number of torus penetrations. The effective pumping speed of the system is 1×10^5 liters/sec, which is sufficient to remove spent gases and impurities from the plasma in 25 sec when evacuating the torus from 10^{-4} torr (pressure after the plasma quench) to 10^{-6} torr. The bundle divertor removes the majority of the helium produced by the fusion reaction from the plasma during the burn; therefore, a pumpdown pressure of 10^{-6} torr is adequate. Should a lower torus pressure ever be required, the pumping system will achieve a pressure of 10^{-8} torr in approximately 60 sec.

The roughing pumps are oil-free mechanical units. In addition to initial pumpdown of the torus, they are used to maintain a forceline pressure of 10^{-2} torr at the high vacuum pumps.

Gases discharged from the mechanical pumps are collected in a pressurized tank and retained for a subsequent tritium retrieval process. Gas collection for the high vacuum pumps is accomplished by regenerating the cryopanels and pumping desorbed gases into a Zr-Al gettering tank. A discussion of the fuel handling system is found in Sect. 20.

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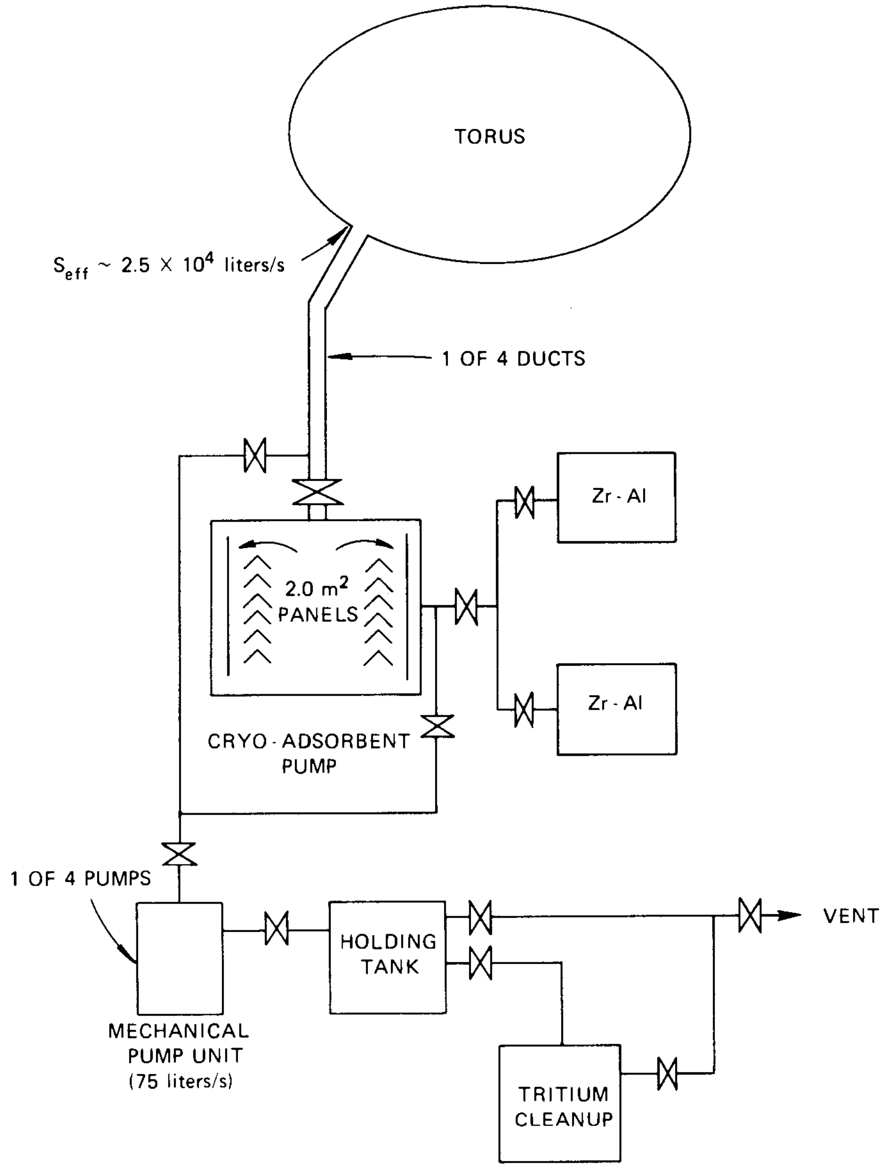

Fig. 17.1. Torus vacuum pumping system.

18. HEAT TRANSPORT SYSTEM

18.1 FUNCTION

The heat transport system (HTS) removes heat from components, transports it outside the facility, and rejects it to the environment via a cooling tower. The system parameters are listed in Table 18.1.

Table 18.1. Heat transport system parameters

Major heat loads[a]	Peak	Average
First wall	113 MW	100 MW
Shield	1224 MW	1090 MW
Divertor energy collector	113 MW	100 MW
Neutral beam lines	100 MW	1 MW
Divertor coil (I^2R)	70 MW	62 MW
PF coil system		
Copper coils	4 MW	4 MW
Power conversion equipment	84 MW	3 MW
TF coil helium refrigerator	17 MW	17 MW
TF coil liquid nitrogen	2 MW	2 MW
Coolant		
First wall	Water	
Shield	Borated water	
Divertor energy collector	Water	
Divertor (I^2R)(copper coils)	Deionized water	
Ion source magnet	Deionized water	

[a]Partial list.

18.2 DESIGN REQUIREMENTS

The HTS will be designed for reliability consistent with requirements for facility availability, incorporating redundant components as necessary to meet these requirements. Radiological safety will be ensured with intermediate coolant loops between the primary coolant loop and the plant circulating water system. Components outside the buffer loop will be designed to handle mean thermal power plus a 50% margin for nonsteady operation. Components requiring scheduled maintenance will be located in areas where contact maintenance is permitted.

18.3 DESIGN DESCRIPTION

The HTS consists of two major subsystems. A water cooling system removes heat from the first wall, the shield, the divertor energy collector, the copper coils, and the beam lines. A cryogenic cooling system removes heat from the superconducting coils and thermal shields located in the dewar, the main vacuum pumping system, and the auxiliary vacuum pumping system in the injectors. Supercritical helium at 4K, provided by a cryogenic refrigerator, is used to cool the superconducting magnets and the vacuum cryopanels.

Final heat dissipation is provided by a cooling tower, shared by both subsystems. A flow diagram of the heat transport system is shown in Fig. 18.1.

Radiological safety is maintained by using a closed buffer loop and intermediate heat exchangers between the activated primary coolant loop and the external circulating water system.

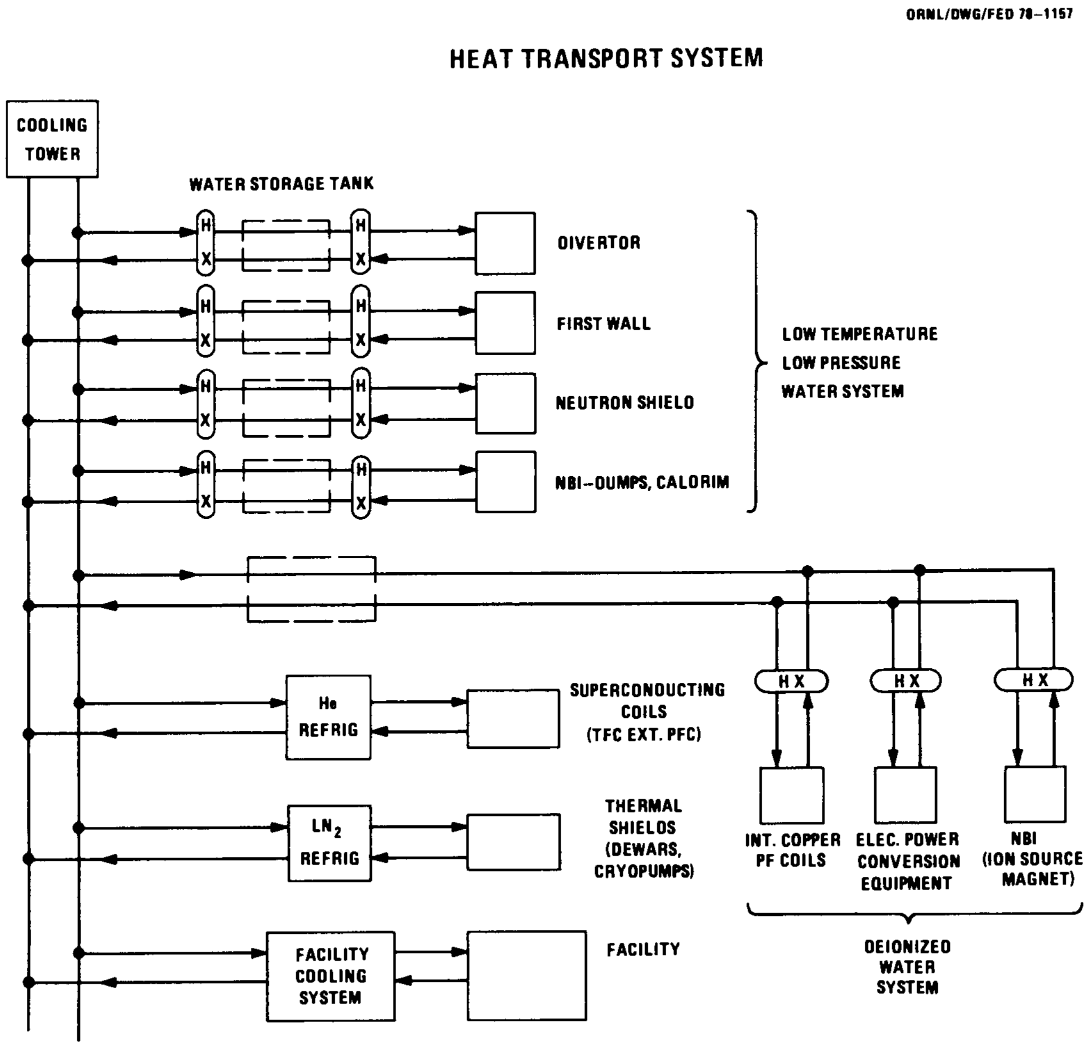

Fig. 18.1. Flow diagram of the heat transport system.

19. FUEL INJECTOR SYSTEM

19.1 FUNCTION

The fuel injector injects D-T fuel in the form of frozen pellets to replenish any fuel lost or consumed during a burn, thus maintaining the plasma characteristics consistent with the fusion process. The pellets must be properly sized and delivered with sufficient velocity to penetrate the hot plasma to an appropriate depth. The fuel injector parameters are given in Table 19.1.

Table 19.1. Fuel injector parameters

Steady-state burn	500 sec
Cycle time	560 sec
Fueling rate (during burn)	0.185 g/sec
Pellet shape	Spherical
Pellet diameter	$\sim$0.5 cm
Pellet velocity	1000-2000 m/sec
Repetition rate	16 pellets/sec

19.2 DESIGN REQUIREMENTS

The injector must be shielded from electrical and magnetic fields in a manner that provides unimpaired operation during steady state or in transient environments. The unit electrical circuits must be protected from fault or induced electrical overloads by adequate fuses or circuit breakers.

Operating temperatures of units of the injector must be controlled by passive or active thermal control systems. The recommended cooling media are liquid nitrogen (LN_2) and liquid helium (LHe).

The injector must have adequate shielding to limit neutron activation of components. Modular designs for components or parts subject to radioactivity or tritium products will provide for ease in maintenance.

There must be no particles from moving mechanical assemblies, lubricants, or outgassing products that could migrate to the vacuum vessel. Vacuum pumps maintain the injector enclosure at a pressure equal to or lower than that of the vacuum chamber and exhaust evaporated D-T products and gaseous products.

19.3 DESIGN DESCRIPTION

The fueling method chosen for the Reference Design is the injection of pellets of solid deuterium and tritium. Recent theoretical studies indicate that pellet penetration may be reduced to about 0.2 times the plasma radius, compared to earlier estimates requiring penetration to the plasma centerline. This may result in reductions of pellet velocity to as low as 1000 m/sec.

The fuel pellet injector mechanism provides the desired injection velocity by exerting centrifugal force on a fuel pellet. A rapidly rotating disc spinning on a vertical axis captures the fuel pellet, which is introduced to the upper face of the disc close to the central hub. The pellet is accelerated along a formed track mounted on the disc's upper face by centrifugal force and leaves the disc through a tube connected to the plasma chamber. The rotational speed of the disc, the disc diameter, the track geometry, and the pellet characteristics govern the ejection velocity and trajectory. Principal subassemblies of the injector mechanism are the rotating disc with its drive, the device that forms the solid (frozen) pellets from deuterium and tritium gases, and the pellet cutter/deliverer that propels the D-T pellet to the impeller.

The entire assembly is housed in a vacuum-tight container. A vacuum-tight tube connects the main unit to the plasma chamber and serves as a clear passage for pellet entrance to the plasma chamber. While the pellets are being injected, the pellet injection device is subjected to direct thermal radiation and neutron streaming. The entire vacuum-tight housing and the interfacing subsystems are shielded from possible radioactivity in the vicinity of the device for personnel protection.

The pellet injector requires these supporting components and subsystems: vacuum pumping, liquid helium supply and plumbing, electrical power, instrumentation and control, basic structure to support injector housing and passive shielding, and filtration and recycling for handling misfired pellets or gaseous tritium products. Figure 19.1 illustrates the functions of the system.

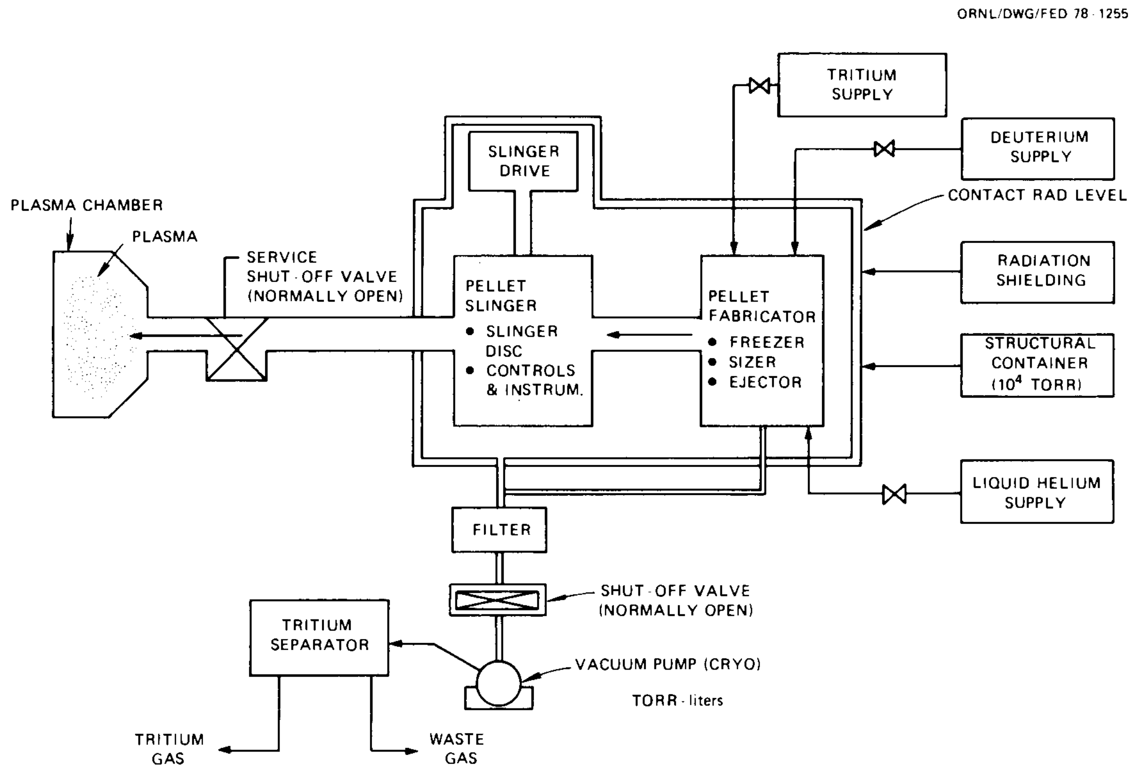

Fig. 19.1. Fuel pellet injector system.

The injector and major subsystem components will be assembled as a single removable unit, with provisions for remote removal and capping of subsystem lines when the pellet assembly is removed for service. The injector will be designed in functional modules that facilitate disassembly and repair. The housing in which moving parts are enclosed must contain any component parts that fail and must protect tritium system and lines from damage or loss of tritium from the injector system.

Two or more pellet injector systems will be installed with one unit always in standby mode. Suitable instrumentation and controls will warn of malfunction in the operating unit and switch the standby unit into operation.

20. FUEL HANDLING SYSTEM

20.1 FUNCTION

The fuel handling system collects tritium and other gases from the shutdown plasma and the divertor, puts these gases in a form suitable for fueling, and provides storage for the fuel. It also provides the means to cope with tritium venting.

The discussion of the fuel processing system is broken down into three functional sections: spent plasma trapping and processing, tritium standby storage and inventory, and tritium containment and cleanup.

20.2 SPENT PLASMA TRAPPING AND PROCESSING

Spent plasma trapping will be accomplished by (1) the torus vacuum compound cryopumps, which use a 4.2K chevron to pump deuterium, tritium, and hydrogen and a 4.2K molecular sieve panel to pump helium, and (2) the liquid lithium droplet cloud in the divertor collection chamber (assuming the lithium rain will pump helium).

Processing and purification of the plasma will be accomplished by combinations of mechanical filtration, gettering, cryogenic trapping, cryogenic distillation, and catalytic cracking-oxidation. Separation of the tritium from the lithium in the divertor will be accomplished primarily by sorption of tritium on solid yttrium or perhaps by a permeation-diffusion process using niobium.

The cryogenic distillation complex has capability to produce adequately pure product streams of D-T, D_2, HD, and T_2; additionally, this complex will accept a side stream from the contaminated deuterium collected on the cryoadsorbent panels of the neutral beam vacuum system for tritium removal. In principle, a pure tritium stream will not be needed if pellets can be made from D-T mixtures.

20.3 TRITIUM STANDBY STORAGE AND INVENTORY

The tritium is stored as solidified tritide on depleted uranium chips in three separate, compartmented, isolated generator storage units. The total on-site tritium inventory will not exceed 10 kg, which is adequate for 28 hr of reactor operation at an 89% duty factor without any reprocessing of the plasma. The minimum inventory depends on the reprocessing rate of the spent plasma.

20.4 TRITIUM CONTAINMENT AND CLEANUP

Each of the three tritium storage-generator units is provided with its own dedicated heaters, coolers, controllers, inert gas blanketing, vault, and emergency protection system. Pumps for tritium processing operations should not have rotating mechanical seals; if this criterion cannot be met, complete containment of the pump in a separate sealed enclosure will be required. All tritium processing equipment is double enclosed, and the volumes enclosed by these jackets are to be continuously flushed by a circulating inert buffer gas processed by a cleanup system to maintain a low cover-gas tritium level.

Tritium cleanup systems are to be provided for the off-gas holding tanks, the vacuum vessel, and the reactor cell; the size of these systems is based on cleanup periods of 120 hr following a major release into the reactor cell and 48 hr for the reactor vessel prior to reactor shutdown and maintenance. Ample storage capacity is provided for all closed-circuit vacuum vessel and shield system aqueous coolants to contain these materials when the system must be drained for equipment maintenance.

21. RADIOACTIVE WASTE HANDLING SYSTEM

21.1 FUNCTION

The purpose of the radioactive waste (radwaste) handling system is to transport and store activated material and components.

21.2 DESIGN DESCRIPTION

The fission reactor industry has a radioactive waste problem, arising largely from fission products, which is more severe than that anticipated for fusion reactors. A well-established radwaste handling technology is available from the fission field, and it is expected that fusion radwaste disposal problems will be dealt with largely through adaptation of this existing technology.

For the Reference Design, contaminated materials requiring disposal consist of portions of the tokamak itself and components which become contaminated through contact with the coolant.

22. EXPERIMENTAL AREA VENTILATION SYSTEM

22.1 FUNCTION

The experimental area ventilation system provides routine ventilation to the reactor containment building and copes with possible tritium venting to the same area.

22.2 DESIGN DESCRIPTION

This system is expected to be based on the results of Tritium Systems Test Assembly (TSTA) developments. Ventilation and tritium cleanup are expected to follow the basic TSTA approach, which consists of circulating air plus tritium through catalytic convertors to form T_2O. This tritiated water will be collected, either through refrigeration or molecular sieves, and then stored.

The major functions of a ventilation system include containment of activated air and tritium leaks. The use of a vacuum building diminishes the activated air problem and is compatible with a tritium cleanup system.

23. INSTRUMENTATION, CONTROL, AND DATA HANDLING SYSTEM

23.1 FUNCTION

The instrumentation, control, and data handling (ICDH) system automatically controls the tokamak plasma and systems through all phases of operation. Semiautonomous operation of subsystem controls and diagnostics is carried out under the supervision of a central computerized control. Data from the control and diagnostics instrumentation are processed and then displayed or stored in an archival system for later on-call display. Fail-safe operations and degraded mode options will receive significant emphasis in the design of the ICDH system.

23.2 DESIGN REQUIREMENTS

The ICDH system will consist of a central operations station and several semiautonomous control substations. The system is centralized in that the machine can be started, operated, and secured from the central operations station and in that the data relevant to machine performance are stored in a central data pool. Upon direct assignment from the central operations station, the semiautonomous control substations may operate a particular subsystem. However, the responsibility of monitoring all operations and the ability to override commands from substations will remain with the central operations station.

Individual subsystem operations, such as testing and checkout, will be prevented from affecting other elements (operating or dormant) by a system of interlocks and overrides. As the environment permits, each control substation will be located in proximity to the equipment it monitors and controls to facilitate testing and checkout.

The ICDH system will be constructed with normal industry practices, modified as necessary for shielding from detrimental effects of nuclear radiation and magnetic fields. The equipment will incorporate redundancy, diversity, and fail-safe features to minimize maintenance needs. Only monitoring devices and communications links and lines will require remote maintenance. With the exception of certain monitoring devices that may require periodic replacement because of exposure to plasma operation, the system has a design life of 10 years.

23.3 DESIGN DESCRIPTION

The ICDH system has four major divisions, each with its own monitoring and data handling.

23.3.1 Operations Control

This subsystem controls all machine operations during all phases of the burn cycle and the immediate preparation for the next burn cycle. The supervisory control and the central operations station form this division.

23.3.2 Instrumentation and Monitoring

This subsystem measures all pertinent variables and processes the data through appropriate algorithms. The instrumentation used for this normal control and monitoring will use multiplexing microwave and light transmission to facilitate maintenance and to reduce the space needed for penetrations to the machine from the operating stations.

It is desirable to monitor the following parameters:

(1) spatial and temporal variations in plasma density,
(2) plasma electron and ion temperatures,
(3) neutron flux at the first wall,
(4) tritium levels at selected locations,
(5) divertor temperatures and coil currents,
(6) poloidal magnetic field strength and coil currents,
(7) toroidal magnetic field strength and coil currents,
(8) neutral beam parameters and status,
(9) rf heating system parameters and status, and
(10) fueling rate.

23.3.3 Auxiliaries Control

This subsystem controls and monitors the elements that are less frequently commanded, including the TF power system, the vacuum pumping systems, the refrigeration systems, and the electrical substation.

23.3.4 Abort Control

The abort control subsystem is a fail-safe, scram-type system that prevents machine damage in the event of any expected disruption or operation anomaly. The instrumentation for this subsystem will receive the strongest emphasis on reliability and redundancy, and variables related to this area will be continually monitored at all stations.

24. ASSEMBLY AND MAINTENANCE

In this section, the design objective for the mechanical arrangement of the Reference Design is reviewed, the requirements derived from this objective are presented, and the maintenance required for each major subsystem is discussed. The equipment needed for remote maintenance is described, and detailed procedures for replacement of a torus sector and a TF coil are given.

24.1 DESIGN OBJECTIVE

In the selection of the mechanical arrangement for the Reference Design, the main objective is a device that exhibits a high degree of maintainability. In a commercial power reactor, where blanket replacement is a regularly scheduled maintenance operation, rapid replacement of torus sectors is a major factor in determining plant availability and hence the cost of electricity. Therefore, the need for rapid sector replacement is derived primarily from the need to develop a design which can be extrapolated to a commercial reactor and secondarily from the need to achieve good maintainability for the Reference Design.

24.2 DESIGN REQUIREMENTS

All major maintenance operations must be performed in the hot cell, in parallel with tokamak operations. To simplify these operations, all components are designed for ease of replacement. In-vessel operations are kept to a minimum and use externally mounted, readily retrievable equipment.

It must be possible to replace the torus sectors without dismantling the dewars to preclude the need for thermal cycling of TF coils.

During a sector replacement operation shear motion between adjacent sectors must be avoided to allow use of overlapping components such as shadow shields. During mating of the replacement sectors, the final increment of motion must be mechanically constrained by either the remote handling equipment or the sector support unit; operator control must not be relied on for fine positioning of torus sectors.

Finally, it must be possible to perform a remote leak check on the torus joint seal.

To meet the design requirements and improve the device maintainability, three major changes to the Baseline Design were incorporated into the evolution of the Reference Design. They are:

(1) a reduction in the number of TF coils, which creates enough room between coils to permit replacement of a torus sector without removal of any coils;

(2) an increase in the TF coil bore, which makes it possible to relocate the PF coils during sector replacement and thus eliminates the need for PF coil segmentation;

(3) the use of mechanical torus joints, made possible by the use of an evacuated reactor cell, which eliminates the need for internal cutting and welding to replace a torus sector.

24.3 MAINTENANCE FUNCTIONS

The major mechanical assemblies of the Reference Design, their functions, and their maintenance requirements are discussed below. Figure 24.1 is a schematic diagram of these assemblies.

24.3.1 Torus

The torus consists of 16 sectors which are bolted together and mechanically sealed. Its weight is supported on four quadrant supports which are independently connected to the ground. When all 16 sectors are joined, a rigid structure is formed which encircles a pair of bucking rings. Electromagnetic centering (radially inward) forces are reacted by the bucking rings. Electromagnetic forces in the radially outward direction are relatively small and are reacted by the intersector bolts.

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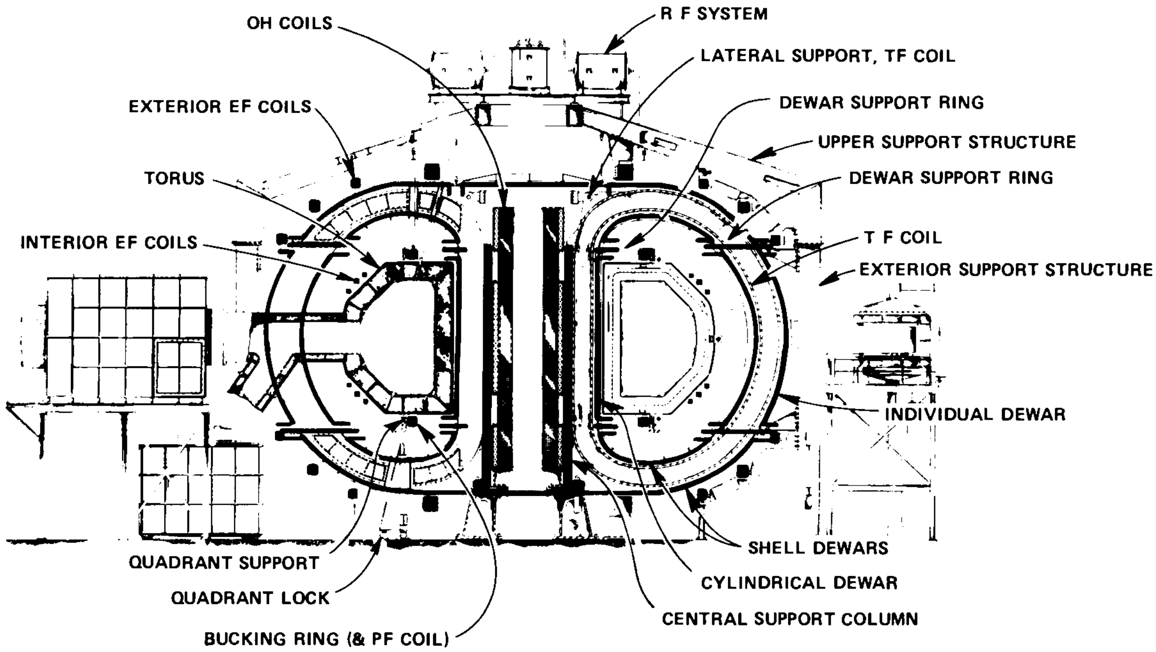

Fig. 24.1. Components involved in the assembly and maintenance operation.

Each quadrant support carries four torus sectors. To remove a sector, one quadrant is moved radially outward a small distance (∿10 cm) by releasing the quadrant support lock and removing the intersector tie rods. The horizontal motion of the quadrant is controlled by the parallelogram arrangement of the support struts. (See Sect. 24.5.1 for complete sector replacement procedure.)

24.3.2 TF Coils

The TF coils are supported by the central support column, the intercoil supports, and, during initial assembly, a lateral support member that supports the imbalance in the TF coil gravity loads. All coil supports are cryogenically cooled. If a coil is removed during maintenance operations, a temporary outboard strut is used to support the remaining coils.

24.3.3 Interior PF Coils

The interior PF coils (within the TF coil bore) are supported from the outer support structure (see Sect. 24.3.7) via the dewar support rings. The six coils closest to the midplane are relocated to a parking position during sector removal. During a TF coil replacement operation, the interior PF coils are segmented.

24.3.4 Exterior PF Coils

The lower exterior PF coils are superconducting coils and are supported from the outer support structure (see Sect. 24.3.7). During a TF coil replacement, they are lowered to a parking location below floor level. Replacement of the lower coils requires coil segmentation and is extremely difficult to perform. Further study is required to identify alternate replacement techniques.

The upper exterior PF coils are fixed to the upper support structure (see Sect. 24.3.8). To replace these coils, the upper support structure is raised and translated to an elevated storage area.

24.3.5 OH Coils

The OH coils are supported on the machine base and constrained laterally by the central support column. Replacement of OH coils requires removal of the rf unit and the center dewar panel.

24.3.6 Dewars

The dewars are supported by the outer and the upper support structures. The individual dewars are accessible from the exterior of the machine. The interior portions of the common dewars are accessible by removal of one or more torus sectors. The upper shell dewar is accessible by relocation of the upper support structure to its parking position. Access to the lower shell dewar requires relocation of the lower exterior PF coils.

24.3.7 Outer Support Structure

The outer support structure supports the exterior PF coils, the interior PF coils, the individual dewars, and the outboard part of the shell dewars. Removal of one column of the outer support structure is required for a TF coil replacement (see Sect. 24.5.2).

24.3.8 Upper Support Structure

The rf unit, the upper exterior PF coils, and the central dewar are supported by the upper support structure. For major maintenance operations, such as a TF coil replacement, the upper support structure is raised and translated to an elevated parking location.

24.4 REMOTE MAINTENANCE EQUIPMENT

Radioactivity levels within the reactor cell dictate that all maintenance operations be performed with remote equipment. Two types of remote maintenance equipment are used: general purpose equipment is used for nonspecific tasks such as servicing coolant and electrical connections and tightening bolts, and special purpose equipment is used for specific and complex tasks such as torus sector replacement or TF coil transport.

24.4.1 General Purpose Equipment

24.4.1.1 Bridge mounted manipulator system

The bridge mounted manipulator system (BMMS) is a shielded, manned unit with a pair of servomanipulators; it services equipment above the torus midplane.

24.4.1.2 Floor mounted manipulator system

The floor mounted manipulator system (FMMS) is a shielded, manned unit used to service equipment below the torus midplane.

24.4.1.3 Floor mounted module

The floor mounted module is an unmanned unit with small servomanipulators for use in floor areas that are inaccessible to the FMMS, for example, the area beneath the torus.

24.4.1.4 Polar crane

The polar crane provides mobility for the BMMS. It is also used during replacement of the OH coils, the rf unit, and the exterior PF coils.

24.4.2 Special Purpose Equipment

24.4.2.1 Sector replacement module

The sector replacement module (SRM) provides all the functions necessary for replacement of a torus sector, including transportation between the hot cell and the reactor cell. When the SRM is positioned at the torus, it is latched to the outer support structure. Two rails extend from the SRM and latch to the quadrant support. A rail mounted unit translates the torus sectors radially into the SRM after first removing the intersector tie rods. Where a sector requires lateral motion prior to radial removal between the coils, an X-Y table mounted on the SRM rails is used.

24.4.2.2 TF coil transport module

The TF coil transport module jacks up the TF coil, removes the lateral support connection and the intercoil connections, and transports the coil to the hot cell.

24.4.2.3 Central bore unit

The central bore unit is carried by the BMMS and extends a boom into the toroidal bore to service the OH coil connections.

24.4.2.4 Injector transport module

The injector transport module (ITM) transports neutral beam injectors and fuel injectors between the hot cell and the reactor cell. It works in conjunction with the BMMS to make and break service lines and duct connections.

24.5 MAINTENANCE OPERATIONS

Two major maintenance operations, a torus sector replacement and a TF coil replacement, are described.

24.5.1 Sector Replacement Procedure

To remove a sector, the entire quadrant is first moved radially outward to provide intersector clearance. As shown schematically in Fig. 24.2, sector A of the quadrant is aligned between two TF coils and may be moved directly out. Removal of sectors B, C, and D requires prior removal of sector A.

The procedure for the removal of sector A of quadrant I is as follows:

(1) Remove injector units and cryopumps in quadrant I.
(2) Disconnect service lines to quadrant I.
(3) Relocate interior PF coils to their parking position.
(4) Transport SRM to sector A.
(5) Latch SRM to the outer support structure.
(6) Extend two lower rails from SRM and latch to the quadrant support rail.
(7) Latch SRM to sector A.
(8) Disconnect torus sector joints AD and DA.
(9) Remove quadrant lock and move quadrant I radially outward.
(10) Disconnect torus sector joint AB.
(11) Jack up sector A to remove weight from quadrant support rail.
(12) Translate sector A radially outward.
(13) Release latches holding SRM to outer support structure and retract SRM rails.

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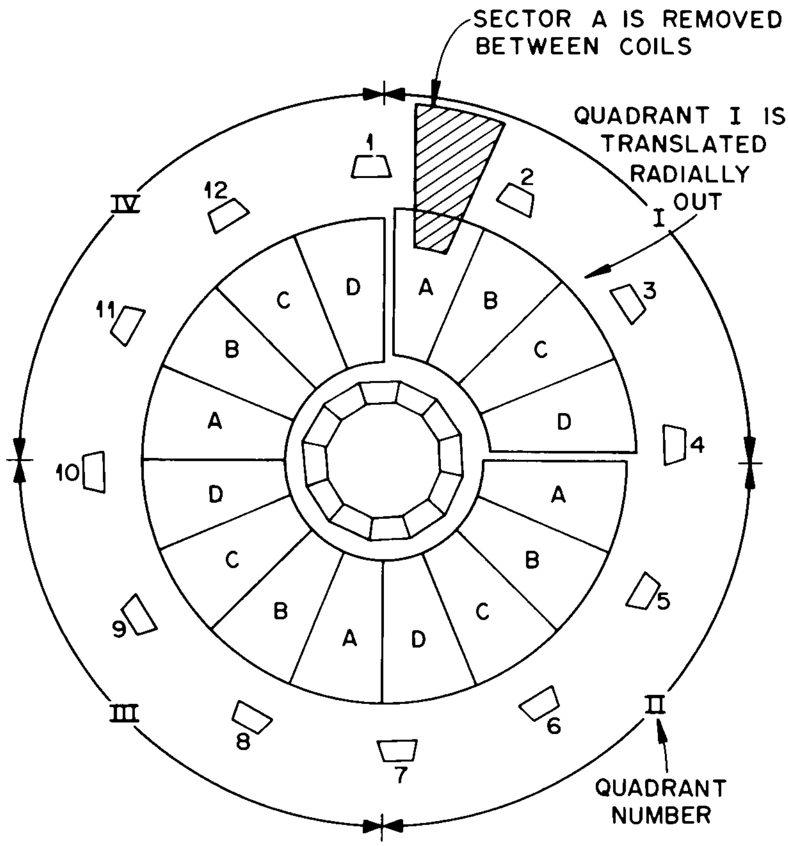
SECTOR REPLACEMENT PROCEDURE



SECTOR	REMOVAL PROCEDURE
A	REMOVE BETWEEN COILS 1 & 2
B	REMOVE SECTOR A, THEN REMOVE B BETWEEN COILS 1 & 2
C	REMOVE SECTOR A; ROTATE B INTO SPACE VACATED BY A; REMOVE C BETWEEN COILS 2 & 3
D	REMOVE SECTOR A, OF QUADRANT II REMOVE D BETWEEN COILS 4 & 5

Fig. 24.2. Torus sector removal procedure.

(14) Transport SRM (with sector A) to the hot cell.

(15) Install a replacement sector, reversing this procedure.

24.5.2 TF Coil Replacement Procedure

(1) Remove the three torus sectors closest to the damaged TF coil.

(2) Remove the upper support structure complete with rf heating unit and PF coils; place in parking location.

(3) Install an outboard strut on the TF coil that is diametrically opposite the damaged TF coil.

(4) Place lower external PF coils in their parking location.

(5) Remove the external support column behind the damaged TF coil.

(6) Remove the individual dewar and a 30° sector of the cylindrical dewar and shell dewars.

(7) Remove a segment of each internal PF coil, the sector support rails, and the bucking rings.

(8) Remove the intercoil support structure.

(9) Position the coil transport module (CTM) at the damaged TF coil.

(10) Latch the TF coil to the CTM.

(11) Jack up the TF coil to remove its weight from the central support column.

(12) Remove the upper shear connection on the lateral support.

(13) Transport TF coil to the hot cell.

(14) Install a replacement TF coil, essentially reversing this procedure.

25. FACILITIES

25.1 FUNCTION

The facilities necessary for the Reference Design include the reactor cell and the support buildings. The reactor cell must provide a 10^{-4}-torr vacuum environment to house the tokamak and all closely coupled auxiliary systems, containment of the device that ensures compliance with regulations concerning radioactivity releases to the environment, radiation shielding as required to reduce dose levels to acceptable values both outside the device and in the basement area (where auxiliary equipment and activities are located), and working space for the various remote systems operations. The support buildings provide working areas and containment for all operations not included in the reactor cell, such as electrical power supplies, control, auxiliary systems, laboratories, offices, and others (i.e., balance of plant). Additionally, this category includes heavy-duty cranes, which are considered to be an integral part of some of the structures. Only the reactor cell is described in detail; the support buildings are conventional. Parameters of the facilities are listed in Table 25.1.

25.2 DESIGN REQUIREMENTS

Two containment vessels make up the reactor cell: an exterior concrete enclosure and an interior metal chamber. The exterior concrete enclosure will be thick enough to support a 1-atm external pressure load and limit the annual radiation dose to the outside environment to less than 100 mrem. The interior metal chamber will be large enough to house the tokamak, allow for maintenance, support a minimum partial pressure to maintain a guard vacuum between the two containment vessels and to provide for leak detection, act as an additional containment vessel for tritium, and provide a safety barrier to contain any projectiles that may be generated during a reactor fault condition. A cross section of the containment building is shown in Fig. 25.1.

Table 25.1. Facilities parameters

Reactor cell	
Structural	
Concrete enclosure inside diameter	56 m
Concrete enclosure height	37 m
Interior metal chamber inside diameter	48 m
Interior metal chamber height	33 m
Access	One 6- by 13-m clear opening
Concrete enclosure material	Reinforced concrete 2 m thick with 0.6-cm-thick steel liner
Interior metal chamber material	Stainless steel or aluminum
Normal vacuum level	20 torr between concrete enclosure and metal chamber, 10^{-4} torr inside metal chamber
Pumping performance	
Mechanical	63,600 cfm
Roots/mechanical	169,600 cfm
Diffusion	2×10^5 liters/sec
Pumpdown time	
From 1 atm to 20 torr	3 hr
From 20 torr to 10^{-3} torr	2 hr
From 10^{-3} torr to 10^{-4} torr	1 hr

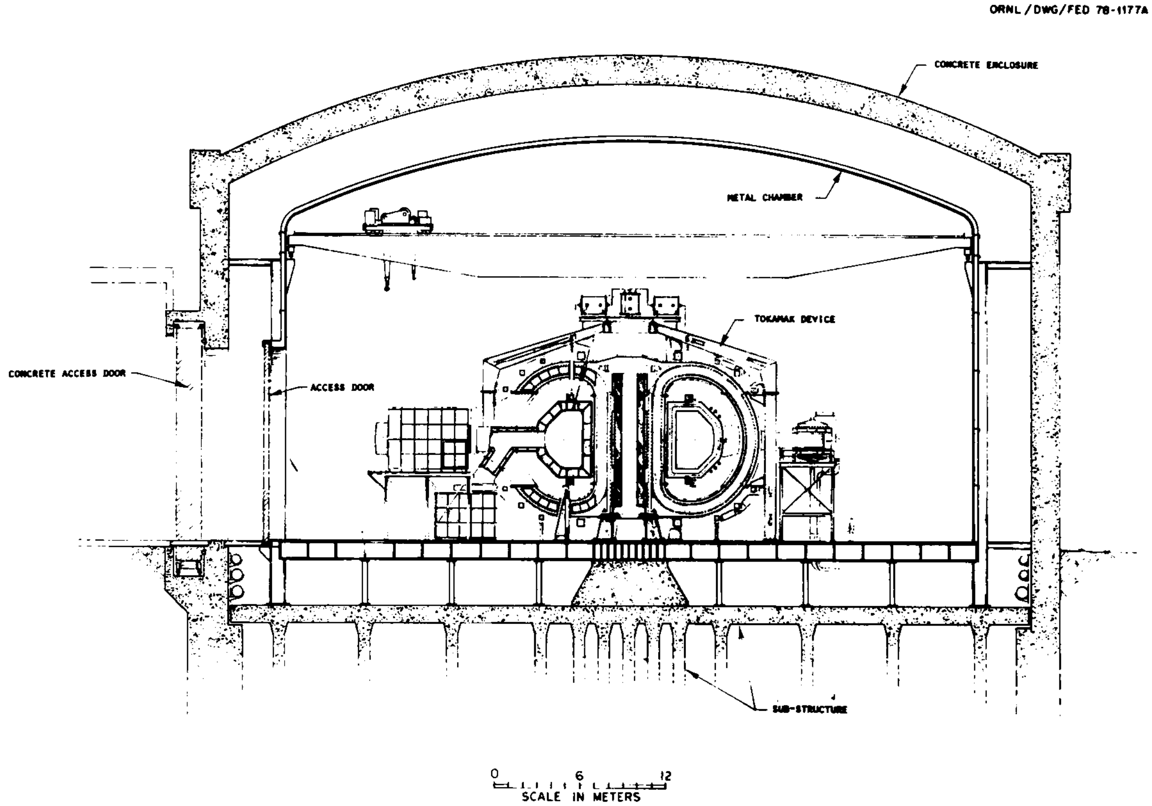


Fig. 25.1. Cross section of reactor cell.

25.3 DESIGN DESCRIPTION

The basic facility consists of a 48-m-diam by 33-m-high vacuum chamber, a shielded assembly/disassembly area, a test control center, an office building, cryogenic and vacuum equipment areas, and associated facility and test support systems. Installation and removal of the tokamak components are facilitated by a 6- by 13-m door that connects the reactor cell to the assembly/disassembly area. Vacuum levels are maintained by a closed vacuum pumping system consisting of diffusion pumps, Roots blowers, and mechanical pumps. Exhaust gases are cycled through a tritium processing system before venting.

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