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TOKAMAK FUSION CORE EXPERIMENT: DESIGN STUDIES BASED ON SUPERCONDUCTING AND HYBRID TOROIDAL FIELD COILS

Design Overview

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FUSION ENGINEERING DESIGN CENTER

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TOKAMAK FUSION CORE EXPERIMENT:
DESIGN STUDIES BASED ON SUPERCONDUCTING
AND HYBRID TOROIDAL FIELD COILS

DESIGN OVERVIEW

Edited by
C. A. Flanagan

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FOREWORD

This overview document summarizes the scoping studies and pre-conceptual design studies performed for the Tokamak Fusion Core Experiment (TFCX) class of device during FY 1983. The studies reported here are for candidate TFCX design options that use superconducting toroidal field (TF) coils. The FY 1984 studies will continue the present effort and will be expanded to include TFCX design options that use resistive copper TF coils.

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CONTRIBUTORS TO TFCX DESIGN STUDIES

FEDC

R. J. Barrett
S. K. Borowski
T. G. Brown
M. K. Bullock
G. T. Bussell
C. A. Flanagan
G. M. Fuller
J. D. Galambos
G. E. Gorker
J. R. Haines
L. M. Hively
R. J. Hooper
S. S. Kalsi
M. H. Kunselman
V. D. Lee

S. K. Makres
D. H. Metzler
J. B. Miller
J. G. Murray
C. E. Nelson
Y-K. M. Peng
R. L. Reid
W. T. Reiersen
K. E. Rothe
T. E. Shannon
P. T. Spampinato
V. C. Srivastava
D. J. Strickler
F. W. Wiffen

ANL

C. C. Baker
P. A. Finn
Y. Gohar
R. F. Mattas

LANL

J. D. Rogers

MIT

D. B. Montgomery
R. J. Thome
J. Schultz

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ABSTRACT

This document is a design overview that describes the scoping studies and preconceptual design effort performed in FY 1983 on the Tokamak Fusion Core Experiment (TFCX) class of device. These studies focussed on devices with all-superconducting toroidal field (TF) coils and on devices with superconducting TF coils supplemented with copper TF coil inserts located in the bore of the TF coils in the shield region. Each class of device is designed to satisfy the mission of ignition and long pulse equilibrium burn. Typical design parameters are: major radius = 3.75 m, minor radius = 1.0 m, field on axis = 4.5 T, plasma current = 7.0 MA. These designs rely on lower hybrid (LHRH) current rampup and heating to ignition using ion cyclotron range of frequency (ICRF). A pumped limiter has been assumed for impurity control. The present document is a design overview; a more detailed design description is contained in a companion document.

1. INTRODUCTION

In the fall of 1982, the Magnetic Fusion Advisory Committee (MFAC) considered numerous upgrades of the Tokamak Fusion Test Reactor (TFTR), including new devices. The Princeton Plasma Physics Laboratory (PPPL) recommended (and the MFAC endorsed) an in-depth study of a superconducting long-pulse ignition device. Accordingly, during FY 1983, engineering studies have been pursued exploring this class of device. The preconceptual studies are being carried out at the Fusion Engineering Design Center (FEDC) and PPPL, with broad community support, to establish the characteristics of the next large tokamak experimental device.

The mission for this device, designated the Tokamak Fusion Core Experiment (TFCX), is to achieve ignition and long-pulse equilibrium burn. To the extent practical, it is desirable to have the device serve as a focus for the development of future fusion technologies. It is important to the project that the cost of the device be minimized. Accordingly, high-fluence nuclear testing has not been incorporated into the mission, and the nuclear testing objectives will be relegated to other future fusion programs to achieve.

Devices with the following toroidal field (TF) coil options are being considered for the TFCX facility: (1) an all-superconducting option (TFCX-S), (2) a copper-superconducting hybrid option (TFCX-H), and (3) an all-copper option (TFCX-C). It is planned that a choice be made following a preconceptual design for these candidates. These preconceptual design studies will be carried to a level that not only ensures design feasibility within the subsystem envelopes developed but also provides a basis for estimating the associated subsystem costs.

The efforts in FY 1983 focussed predominantly on the all-superconducting option and on the copper-superconducting hybrid option. For these two options, design studies were completed to establish the overall characteristics of each class of device to establish the design definition of all of the major subsystems, and to develop associated capital cost projections. This paper describes the TFCX studies performed in FY 1983. The studies will continue in FY 1984 with the focus on developing all-copper TF coil options. It is also intended to make a selection of

options in mid-FY 1984 and to initiate a conceptual design using the results to date as a point of departure.

It should be noted that no decision has been made on a final design for any of the three options and considerable flexibility exists in establishing the final size and associated performance parameters for each option. Included in establishing the final parameters and size will be evaluations and decisions on the physics basis to be used. This will be done in FY 1984. However, for purpose of discussion of key features of each option, nominal cases are presented in this paper for the options studied to date.

Section 2 discusses issues and options of interest to TFCX. Section 2 discusses physics considerations. Section 4 presents a description of configuration features. Section 5 discusses the plasma operating scenario consideration. Section 6 summarizes numerous systems studies performed using the tokamak systems code. Sections 7 through 11 describe, respectively, the magnetic, nuclear, plasma heating, electrical, and auxiliary systems. Section 12 discusses the facilities considerations.

Background information leading to these TFCX design studies can be found in Ref. 1. A more detailed design description, together with the supporting trade studies, can be found in Ref. 2.

2. ISSUES AND OPTIONS

The TFCX (along with complementary, nonfusion facilities) must address the remaining issues requiring resolution before a final commitment is made to an Engineering Test Reactor (ETR). The physics and technology issues to be addressed by TFCX will be those remaining after the operation of the Doublet III Upgrade (DIII-U), Fusion Test Reactor (TFTR), and the Joint European Torus (JET).

Most of the issues addressed by TFCX will be related to ignition and/or long-pulse operation. Information will be required on post-ignition confinement and burn control, on maintaining a high-beta configuration for a period which is long compared with magnetic diffusion times, on impurity control and power/particle handling at long pulse and with a reactor-like core plasma, on initiating plasma currents using noninductive current drive, and on ash accumulation and removal. Closely related technology issues are testing of ion cyclotron and lower hybrid launcher concepts designed for the radiation environment, for tritium containment, and for maintainability and management of disruptive energy dumps at more nearly reactor-relevant levels (5 to 10 MA and ≥ 100 MJ).

The next tokamak facility, along with all advanced fusion devices, must provide information on component and system reliability and availability. Reactor-relevant TF coils must be employed on operating tokamaks; however, no existing or planned U.S. tokamak uses superconducting coils. Enlightenment on maintenance is likely to come exclusively from operating devices. Before a next tokamak, only TFTR must face up to remote maintenance, and unfortunately, the TFTR configuration is unsuitable for reactor use. Thus, remote maintenance as well as component and system reliability will be important issues addressed by TFCX. Other systems-related issues are tritium management and the development of diagnostics, data acquisition, and control systems for the full fusion environment, including ignition and long-pulse burn.

Most of the issues to be resolved by the next tokamak facility are associated with ignition and long-pulse burn. Therefore, there are three preferred options for the next tokamak facility that can provide

ignition and long-pulse burn capability. These three options are distinguished by the type of TF magnet employed: all-superconductor, a copper-superconducting combination, or all-copper.

The TFCX option with a superconducting TF coil is clearly relevant to future reactor development. However, neutron shielding is required to reduce the nuclear heating in the superconducting coil even for low-fluence operation, and this takes up space and tends to increase the major radius. In addition, the maximum reliable operating field realizable with superconducting coils is substantially less than that for copper coils. The lower field results in an increase in device size for a given ignition parameter. The increased plasma major radius and the aspect ratio associated with superconducting coils are in the direction to incur a potential cost penalty relative to the copper coil option.

Using copper-superconductor combination TF coils provides a potentially attractive hybrid option. Copper TF coils located inside the superconducting TF coils provide shielding for the superconducting coils and higher central fields, resulting in smaller size and potentially lower cost than the all-superconductor option. This option's advantage over the all-superconductor approach is the higher central field at the same or lower cost. The advantage over the all-copper option is its increased relevance to the ETR issues: it does develop superconducting coils and its design is based on these coils. Disadvantages relative to the all-copper options are a somewhat longer schedule and increased complexity.

Using all-copper TF coils removes the superconducting design requirement for shielding the TF coils, potentially reducing size and complexity, and allows higher central fields, allowing an additional decrease in size for equivalent performance. In addition, the need for superconducting coil research and development is eliminated. The principal advantages of the all-copper device are the technical simplicity, reliability, and a shorter construction schedule. The principal disadvantage arises directly from the advantages: the copper coils, the attendant simplicity, and the lack of an accompanying superconducting coil development causes this option to be least relevant to the unresolved superconducting issues of an ETR.

3. PHYSICS CONSIDERATIONS

The overall mission for TFCX dictates that the physics requirements for the device be compatible with a long-pulse, ignited discharge. The key considerations in assessing the prospects for ignition are the quality of confinement and the limits on plasma pressure (beta limits). During initial studies, conventional plasma shapes (D shapes) and associated modest betas have been assumed.

Experiments on present devices indicate a soft beta limit, which may provide a stable operating point during burn. However, some method of varying the operating point (such as varying plasma current, ripple, or impurities) may be required.

The early Fusion Engineering Device (FED) and International Tokamak Reactor (INTOR) design studies were carried out using Alcator scaling for energy confinement and using beta limits roughly consistent with ideal magnetohydrodynamic (MHD) theory. A useful approximate form for the beta limit, deduced from a myriad of low-q MHD stability code runs and associated heating experiments, is

$$\langle \beta \rangle_{\max} = 0.2 \frac{a}{Rq} \frac{1 + \kappa^2}{2} ,$$

where $\langle \beta \rangle$ is the average beta, a is the plasma minor radius, κ is the plasma elongation, R is the major radius, and q is the safety factor. The observation in Doublet III in 1983 of a stable, elongated discharge with a beta of 4.5% is consistent with this formulation.

With regard to confinement, the results of a number of auxiliary heating experiments are in conflict with Alcator scaling. A better fit to such data is

$$\tau_E \propto I_p \text{ (MA)} a \text{ (m)} f(\kappa) ,$$

where τ_E is the energy confinement time, I_p is the plasma current, a is the plasma minor radius, and $f(\kappa)$ is a modifying coefficient that depends on plasma geometry. The size scaling is subject to considerable uncertainty.

The ignition parameter I is defined as the ratio of the alpha heating power P_α to the plasma loss power P_{Loss} . Since $P_\alpha \propto \beta^2 E^4$ and $P_{\text{Loss}} \propto \beta B^2 / \tau_E$, one finds using the relationship

$$q \propto \frac{B_T}{I_p} \frac{1}{R} \propto f(\kappa)$$

that the above equations can be combined to obtain a simple form for the ignition parameter:

$$I = \frac{P_\alpha}{P_{\text{loss}}} \propto I_p^2 B_T.$$

When one is this close to the beta limit, there is an observed dependence of confinement on beta which must be taken into account when estimating the ignition parameter.

A single-null poloidal divertor configuration was developed as part of the INTOR design studies with minimum impact on the poloidal field (PF) coil design and mechanical configuration. This configuration can be accommodated within about the same envelope required for a pumped limiter placed at the bottom of the plasma chamber. Due to the reasonable compatibility of these two configurations, the TFCX preconceptual design studies and the associated option selection have been designed with a pumped limiter. Ongoing divertor and limiter experiments will form the basis for selecting the impurity control option during the conceptual design.

Current ramp-up at low density with rf heating near the lower hybrid frequency [lower hybrid resonant heating (LHRH)] has been successful in several recent tokamak experiments. This technique will be used with a modest ohmic heating (OH) voltage to ramp the plasma current during startup. This will alleviate the inductive volt-second requirement for startup and conserve the volt-seconds for use during the burn. The studies performed in FY 1983 assume a 300-s ignited burn. About 10 MW of LHRH appears sufficient for current ramp-up purposes. The remaining 20 MW required for heating to ignition is provided with ion cyclotron range of frequency (ICRF) heating. A modest amount (0.1 to 1 MW) of

electron cyclotron resonance heating (ECRH) may be required for preionization and has been incorporated.

Some aspects of the physics basis for the TFCX will require development and qualification on existing experiments. High-power heating experiments on TFTR, JET, DIII-D, and the Poloidal Beta Experiment (PBX) will confirm the choice of the transport coefficients and beta limits used for the TFCX design. Initial high-power heating experiments on TFTR and JET are well timed to fit the TFCX needs. These experiments will provide confinement data as well as the physics information necessary for the choice of the impurity control option for the TFCX. If TFTR and JET operate successfully without divertors, then impurity control with the pumped limiter on the TFCX will be the logical choice. If a divertor proves necessary, then additional information on divertor design will be needed and will depend mainly on experiments on Axisymmetric Divertor Experiment-Upgrade (ASDEX-U) at Garching.

The present TFCX design options rely heavily on lower hybrid current drive (LHCD) current ramp-up to reduce the volt-second requirements and associated central solenoid size. This technique shows great promise; however, further development of the LHCD data base will be required. Lower hybrid current ramp-up is being studied on the Princeton Large Torus (PLT) and Alcator C.

The TFCX project will require qualification of ICRF for heating to ignition. Programs are in place to provide the development of rf launchers and the qualification of ICRF necessary to proceed with the TFCX and will test a range of candidate launcher concepts for the TFCX design. The plasma heating efficiency and impurity control problems associated with ICRF heating will continue to be tested on PLT.

A portion of the TFTR experimental program will be devoted to the characterization and understanding of high plasma current disruptions. This information, coupled with an improved data base related to low-voltage startup, would facilitate the design of the TFCX vacuum vessel.

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1. Panel III Report to the Magnetic Fusion Advisory Committee on Possible TFTR Upgrade Options, January 10, 1983.
2. Tokamak Fusion Core Experiment: Design Studies Based on Superconducting and Hybrid Toroidal Field Coils, ORNL/FEDC-83/13, ORNL, FEDC, to be published.

4. CONFIGURATION FEATURES

Two candidate options were selected for preconceptual design during FY 1983. These include an all-superconducting TF coil configuration (TFCX-S) and a hybrid superconducting-copper TF coil system with water-cooled coils located in the shield region in the plane of the TF coils. The removable torus sectors between the TF coils contain no copper coils (TFCX-H).

To provide for practical assembly and maintenance, a number of principles have evolved from the early reactor studies that have significantly influenced the current tokamak configuration. The first is that of accessibility: all systems and subsystems must be designed so that easy access is provided to the device and to each major system within the device. A second guiding principle is the concept of modularity: the major systems and components of the device are assembled from modular pieces. Together, these two items — accessibility and modularity — can lead to design approaches for the tokamak that provide the practical capability to service and repair all systems and components. In the design process, it is essential that important trade-off studies are performed. It is now widely recognized that the tokamak configuration must be designed from the outset to be maintained; it is equally important that all of the many key systems be designed as integrated pieces of the total reactor. The systems integration aspect of the design cannot be overlooked. There are a few key systems that dominate the tokamak configuration, as indicated in Table 4.1. It is primarily these systems that must be designed for practical maintenance.

Table 4.1. Key systems that dominate tokamak configuration designs

TF coils
PF coils
Magnet systems structure
Torus structure
Vacuum systems
Impurity control systems

A third principle is the desirability of providing hands-on maintenance capability at the device perimeter within a day following shutdown. Hands-on maintenance capability at the shield boundary is presently provided by designing the shielding to limit the dose rate to 2.5 mrem/h approximately one day after shutdown. These contact operations are valuable for visual inspections, troubleshooting, equipment setup, and calibration, minor repairs, and supervision of equipment installation or modifications.

In designing the tokamak configuration to achieve the desired goal and satisfy the preceding principles, the key systems of Table 4.1 dominate the design and have received considerable design emphasis in recent tokamak reactor studies. A comparison of the configuration features from a number of recent fusion reactor studies performed by many different design teams (including international contributions through INTOR) suggest a number of desirable engineering features. These desirable features represent the present best tokamak design configuration, not only for a power reactor, but also for the next-generation tokamak.

These desirable configuration features have been incorporated into the TFCX design, including:

- Relatively few superconducting TF coils. (Designs with 12 to 16 coils are being assessed; since the major radius of these designs is smaller than for the FED/INTOR class, the edge ripple considerations tend toward a slightly larger number of TF coils.)
- A torus consisting of modular sectors radially inserted between the outboard legs of the TF coils
- All PF coils located external to the TF coil bore and accessible
- A single, combined vacuum boundary between the plasma chamber and the superconducting TF coils
- Horizontal access at the midplane of the device
- Simplified gravity support for the torus shield and vacuum system

An elevation view of TFCX-S is given in Fig. 4.1 and a plan view is given in Fig. 4.2. An elevation view of TFCX-H is given in Fig. 4.3.

In this design, the entire blanket and first wall, together with essentially all of the bulk shield, can be removed by individual extraction of each torus sector using simple, linear, radial motion. Each torus sector is designed to be highly modular with plug-in auxiliaries, such as heating, testing, and impurity control. A combined vacuum boundary is employed in this design; one side of the boundary is the plasma, the other is the magnet cryostat. The vacuum boundary for the removable shield sector is at the outer surface of the sector, which is sealed to the cryostat after the sector is inserted into the torus. The PF coils are located to permit removal for maintenance by simple vertical motion. Each of the TF coils can be removed for repair or replacement, if necessary. A TF coil is removed by removing the associated torus sectors, vertically removing the vessel dome and the upper PF coils, laterally moving the selected TF coil radially outward a small distance, and then vertically lifting the TF coil out of the device.

Key parameters for a baseline all-superconducting TF design (designated TFCX-S) and a baseline superconducting-hybrid TF design (designated TFCX-H) are given in Table 4.2. Since no selection will be made of the preferred TFCX option until FY 1984, the studies have examined variations from the baseline cases. The two baseline cases given have been used as the benchmark for system and component design studies. However, until the total preconceptual design effort is completed, many of these variations still represent viable candidate design options.

The TFCX-S baseline design is ignited and employs superconducting TF coils with a maximum field at the winding of 10 T. The device has a major radius of 3.75 m, a minor radius of 1.07 m, a plasma current of 7.7 MA, a fusion power of 230 MW, a neutron wall loading of 0.9 MW/m^2 , and sufficient volt-seconds to provide 300 s of burn.

The TFCX-H baseline design is ignited and employs superconducting TF coils with a maximum field at the winding of 8 T complemented by water-cooled copper coils located in the shield region to enhance the field on-axis. The device has a major radius of 3.60 m, a minor radius

Fig. 4.1. TFCX-S elevation view.

Fig. 4.2. TFCX-S plan view.

Fig. 4.3. Elevation and plan view schematics of TFCX-H.

Table 4.2. Key parameters for TFCX-S and TFCX-H

Parameter	TFCX-S	TFCX-H
Major radius, R (m)	3.75	3.60
Plasma radius, a (m)	1.07	0.97
Elongation, κ	1.6	1.6
Triangularity, δ	0.3	0.3
Aspect ratio	3.5	3.7
Average ion temperature, $\langle T_i \rangle$ (keV)	13.0	13.0
Average electron temperature, $\langle T_e \rangle$ (keV)	13.0	13.0
Average electron density, $\langle n_e \rangle$ (10^{20} m^{-3})	1.9	1.9
Maximum field at s/c winding, B_{max} (T)	10	8
Field on-axis, T_t	4.3	4.8
Plasma current, I_p (MA)	7.7	4.8
Safety factor, q_ψ	2.0	2.0
Beta, β (%)	5.9	5.6
Fusion power (MW)	230	250
Neutron wall loading ($\text{MW} \cdot \text{m}^2$)	0.9	1.1
Ignition parameter	1.0	1.0

of 0.97 m, a plasma current of 7.2 MA, a fusion power of 250 MW, a neutron wall loading of 1.1 MW/m^2 , and sufficient volt-seconds to provide 300 s of burn.

5. PLASMA OPERATING SCENARIO

The plasma operating scenario involves the sequence of operations to initiate and execute a long-pulse discharge cycle in TFCX. Since one of the major objectives of TFCX is to explore the physics and engineering of long-pulse equilibrium burn conditions, means of obtaining long-pulse burn are being investigated both theoretically and experimentally.

Earlier designs, such as FED/INTOR, which were based on inductive current startup and maintenance, had the plasma burn phase limited by the poloidal flux capability. In FED, only about 5% of the total flux capability was available to maintain the plasma current for the 100-s burn; most of the total flux capability was expended in establishing the plasma current. The pulsed nature of an inductively driven tokamak tends to impose serious limitations on the lifetime, availability, and economy of a tokamak reactor. Therefore, it is highly desirable to achieve a significant level of noninductive current drive in TFCX.

Recent demonstrations of LHCD have been successful in PLT, Alcator-C, Versator II, JIPPT-II, etc., at modest plasma densities and support the proposed current-drive scenario. A successful radio frequency (rf) current drive at low density does not necessarily imply an indefinite burn phase at the required density (about 10^{14} cm^{-3}). However, such capability will allow application of a large fraction of the OH induction flux capability to the burn phase at high density. This will facilitate for long-burn-pulse capability while reducing the size of the OH solenoid.

A reference plasma operating scenario based on LHCD has been developed for TFCX. Table 5.1 summarizes the scenario.

A typical TFCX plasma discharge cycle involves prefill and preionization, current initiation and current drive to full current, bulk heating, burn, plasma shutdown, and pumpdown. The key features of these phases are discussed below with an emphasis on what happens in each phase to better understand the context of the design.

Table 5.1. TFCX plasma operating scenario
(D-T operation)

Phase	Interval (s)	Heating/fueling requirements	Plasma characteristics			
			Density ($\times 10^{13} \text{ cm}^{-3}$)	Temperature (keV)	Current (MA)	Power (MW)
Prefill chamber		Pressure $> 10^{-5}$ torr from $\sim 10^{-6}$ torr				
Current initiation	0 to ~ 1.0	ECRH power ~ 1 MW	~ 1	~ 0.1	0	0
Current ramp-up	~ 1 to 30	LHRH power ~ 12 MW Fuel	~ 3	~ 1.0	~ 6	0
Bulk heating	30 to 40	ICRH power ≤ 40 MW Fuel Pump limiter	~ 12	13.0	7	~ 250
Burn	40 to 340	ICRH power off Fuel Pump limiter	~ 12	13.0	7	~ 250
Shutdown	340 to 360	Fuel off Particle exhaust	~ 2	~ 1.0	0	0
Pumpdown	360 to 390	Pumpdown pressure				

5.1 PREFILL AND IONIZATION

About 1 MW of extraordinary mode, electron cyclotron resonance frequency wave at about 100 to 110 GHz is launched from the high-field side for a duration of ~ 1 s to preionize and heat the electrons. The heated electrons ($T_e \sim 100$ eV, $n_e \sim 1 \times 10^{13} \text{ cm}^{-3}$) are expected to be located over a radial width of ~ 0.4 m. This permits the application of a low toroidal loop voltage ($V_1 < 25$ V) to initiate and rampup the plasma current.

To effect proper penetration and absorption of the wave, it must come from the high-field side of the plasma with the extraordinary mode, which means the electric field is perpendicular to the toroidal field. An injection angle of 85° is specified but needs substantiation when the exact startup plasma parameters are known.

5.2 CURRENT INITIATION AND RAMP-UP

The baseline operating scenario calls for an all rf startup - saving the OH transformer volt-seconds for the long-pulse burn. Current-drive techniques presently in use indicate a decrease in efficiency (amperes of plasma current driven to watts of LHRH power) as the density is increased. Hence, LHRH is used during the low-density parts of the startup and into the bulk heating period until the density reaches a level where LHRH becomes less effective; the OH transformer is then switched on.

Either D, H, or D-T gas can be used with current drive. For ignited conditions in D-T operation, a startup time of ~ 30 s is used in the baseline scenario. To size the LHRH system for current drive, a range of 0.5 to 7.2 MA was used resulting in a power requirement of about 12 to 14 MW. This power level should be controllable over a 100-dB range. The frequency is twice the lower hybrid frequency to avoid ion interaction. The frequency is ~ 2.7 GHz.

5.3 BULK HEATING

About 40 MW of ICRF power is applied at a frequency of about 70 MHz for about 10 s. The plasma temperature increases from ~ 1 to ~ 13 keV

and plasma density increases to a full density of $\sim 1.2 \times 10^{14} \text{ cm}^{-3}$ at full plasma current of $\sim 7 \text{ MA}$ via pellet fueling and gas puffing, maintaining nearly equal deuterium and tritium content. The contact between plasma and limiter is controlled to achieve adequate particle exhaust through the limiter channels without overheating the leading edge of the pump limiter. The value of q_ψ at the plasma edge is maintained at 2.0 during this phase. Under the given plasma beta and confinement assumptions, ignition is achieved and the fusion power is expected to reach $\sim 250 \text{ MW}$.

5.4 BURN

Once the full plasma conditions of temperature, density, and current are achieved, ignition is obtained and an operating point established. The OH transformer flux capability is then used to maintain the plasma current during the 300-s burn. The pump limiter is assumed to exhaust the helium and control the plasma edge adequately, but auxiliary impurity control or reduction schemes may be needed. An attractive possibility for use in conjunction with a pump limiter is the maintenance of a radiation-cooled plasma edge, which is expected to minimize impurity production and ingestion at the plasma edge. Assuming adequate helium exhaust and impurity control, the plasma burn time is limited either by the volt-second capability of the OH coils or by significant resistive diffusion of the plasma poloidal flux. With nearly classical toroidal plasma conductivity, the latter time scale is estimated to be a few hundred seconds.

An option at the end of the burn would be to recharge the OH transformer, reduce the temperature and density of the plasma, and maintain the current at or near its full value with the LHRH system and then to initiate another 300-s burn pulse by applying the ion cyclotron resonic heating (ICRH) to raise the plasma temperature, bring up the plasma density, and use the OH transformer flux capability to obtain another 300-s burn.

6. SYSTEMS STUDIES

During the last year, a large variety of potential concepts for what the TFCX device should be have been examined from a global systems point of view. These studies tend to be ongoing and rely heavily on the use of the FEDC systems code. This code has been developed over the last five years. It is used to calculate the tokamak configuration, performance, and cost as a function of plasma and engineering parameters. The code has a modular structure to allow independent modeling of each major tokamak system or component. Each module incorporates the latest design thinking in the representation of each major system or component.

The more recent systems studies have coupled the systems code to available MHD and magnetic field strength analysis codes in order to ensure that plasma and magnet systems constraints are satisfied.

The adequacy of the PF system (based on the constraints of plasma shape, burn time, maximum field at the PF coil, and configurational constraints) was determined by iterative execution of the FEDC systems code, an MHD equilibrium code, and EFFI (a code to evaluate magnetic field strength). Figure 6.1 is a flow diagram of the procedure.

The following sections present results of a number of the major systems studies performed during this period.

6.1 DRIVEN-CURRENT TOKAMAK (DCT) SCOPING STUDIES

A series of superconducting Driven-Current Tokamak (DCT) devices were examined early in FY 1983 by FEDC and by PPPL. This series of devices range from a PLT-size plasma device (denoted DCT-2) through an ignition-size D-T device (denoted DCT-8). The smallest device is distinguished by the choice of a TF coil size comparable to that of the Large Coil Test Facility (LCTF). The largest device is about 70% the size of FED.

The DCT-2 device is designed solely for long-pulse hydrogen plasma operation and involves no D-T plasma operation. As such, the role of DCT-2 in the fusion program would be to develop the nonnuclear technology and physics required for an Engineering Test Reactor (ETR).

Fig. 6.1. Flow diagram of system study procedure.

The DCT-8 is an ignited, long-pulse D-T device. The distinctive feature of one option for DCT-8 is to use a TF coil system in which both superconducting and copper coils are used; this facilitates high-field, high-performance operation at reduced size and cost. The objectives for this device combine the long-pulse objectives of a superconducting tokamak with the nuclear technology and physics objectives of a D-T burning experiment. It should be noted that the designation of DCT-8 was changed to TFCX-H later in the year.

The major device parameters for the DCT-2 and DCT-8 are listed in Table 6.1. These parameters are representative of only one design point in the available design space for each device; trade-offs in many of these parameters typically occur during the preconceptual design to identify the cost/performance sensitivities and thereby permit iteration to a final set of parameters.

6.2 DCT-8 CONFIGURATION TRADE STUDY

Four hybrid configuration options were identified as possible candidate designs for the DCT-8 device (see Fig. 6.2). Hybrid configuration options included locating copper TF coils: (1) in the shield, both under the TF coil and in the removable shield module; (2) in the shield region, solely under the TF coil; (3) solely in the removable shield module; and (4) as an integrated entity. An all-superconducting option (Case 1 $\equiv$ TFCX-S) was specified as the candidate option for comparison with the selected hybrid option.

A scoping study was made to reduce the candidate hybrid options down to one approach to compare it with the all-superconducting design. In evaluating the hybrid options, locating a liquid-nitrogen-cooled copper coil with the superconducting TF coil (Case 5) was found to raise the overall reactor cost when compared with Cases 2 through 4. This resulted from the lower-field contribution of the copper, being located further from the plasma, and the inability of the copper/superconducting arrangement to be configured with a variable inboard-to-outboard cross section. Support for the copper coils in the Case 2 arrangement appeared to be quite difficult; therefore, it was dropped as a candidate hybrid

Table 6.1. Key parameters for DCT-2 and DCT-8

	DCT-2	DCT-8
Major radius (m)	2.9	3.6
Plasma radius (m)	0.77	0.75
Plasma elongation	1.6	1.6
Fusion power (MW)	0	390
Neutron wall loading (MW/m ²)	0	2.2
Heating (MW)		
ICRH	33	22
LCRH	0	10
Amplification factor, Q	0	00
Average total beta (%)	3.0	4.3
Plasma current (MA)	4.0	5.9
TF coil clear bore, width × height (m)	2.6 × 3.6	3.6 × 5.2
Cu coil clear bore, width × height (m)	NA	3.0 × 3.9
Field on-axis (T)	4.7	7.0

CASE 2 - Cu IN SHIELD UNDER TF
COIL AND IN REMOVABLE
MODULE

CASE 3 - Cu IN SHIELD UNDER TF
COIL

CASE 4 - Cu IN SHIELD IN
REMOVABLE MODULE

CASE 5 - Cu WITH S/C TF COIL

Fig. 6 2. DCT configuration options.

Locating the copper coils entirely in the shield (Case 4) offered better access to maintain the copper coils and to supply cooling and electrical hookup compared to Case 3; however, supporting the copper coils against their out-of-plane forces appeared to be more difficult. A possible support method for Case 4 was to provide external reinforcement at the outside of the vessel. The support method for Case 3 occurs entirely inside the shield, independent of the removable shield module. The intercoil support structure is attached at the point where the out-of-plane forces appeared to be more difficult. A possible support method for Case 4 was to provide external reinforcement at the outside of the vessel. The support method for Case 3 occurs entirely inside the shield, independent of the removable shield module. The intercoil support structure is attached at the point where the out-of-plane magnetic loads are being applied, rather than relying on the removable shield module to carry the loads in torsion to an external support structure. For this reason, Case 3 was selected as the hybrid copper/superconducting option for further design definition and, ultimately, to be compared with an all-superconducting option (Case 1) and to an all-copper device. (Note: Case 3 of this series ultimately was designated TFCX-H and Case 1 became TFCX-S.)

6.3 SYSTEMS RESULTS FOR TFCX-S AND TFCX-H

System studies performed on a self-consistent basis require

- a well-defined configuration concept,
- specified design guidelines and criteria from which design points can be derived, and
- reasonable models by which performance and operating characteristics can be assessed.

The TFCX-S and TFCX-H design points were derived based upon the design guidelines and criteria listed in Table 6.2. Other than the field enhancement due to the copper insert, the design criteria for the reference designs differ only in the TF coil technology selected for each device.

Table 6.2. TFCX design guidelines and criteria

Parameter	Value
Elongation	1.6
Triangularity	0.3
Safety factor	2.0
Ignition parameter	1.0
Maximum burn time ^a (s)	300
Cumulative burn time (s)	2×10^5
Maximum TF ripple ^b (%)	1.5
Plasma temperature (keV)	13
Startup current maintenance	LHRH
Maximum field at PF coils (T)	8
Maximum field at TF coils (T)	8-10
Winding cavity current density in TF coils (A/cm ²)	2600-2000
Field enhancement on-axis (T)	0-1.6
Allowable TF dose to insulator (rad)	10^9
Allowable TF nuclear heating (mW/cm ³)	
Inboard	1.00
Outboard	0.25
Shutdown dose rate (mrem/h)	1.0
Cooldown time (h)	24

^aInductive driven.

^bJudicious placement of iron for ripple reduction allowed.

There are several distinctions between the early DCT scoping studies and the more recent follow-on system studies. When the early DCT studies were conducted, a well-defined, viable hybrid configuration concept had not been developed. Thus, the distinctions between hybrid and superconducting configurations were not reflected in the DCT study results. In the interim between the two efforts, the FEDC systems code was substantially upgraded and updated. For the more recent studies, the procedure illustrated in Fig. 6.1 was used. The procedure involves iterative execution of three major codes: the FEDC systems and MHD equilibrium codes and EFFI, a code to evaluate magnetic field strength. All three codes are required to ensure that the constraints of the plasma shape, burn time, and maximum field at the PF coils are simultaneously met.

Using the design guidelines and criteria (Table 6.2) and the indicated methodology (Fig. 6.1), design points were established (1) for two configurations using only superconducting TF coils (at maximum field strengths of 8 and 10 T, respectively) and (2) for two configurations using superconducting TF coils supplemented with copper TF coil inserts in the shield sectors (with maximum field strengths of 8 or 10 T, respectively, for the superconducting coils). The key parameters are given for these four cases in Table 6.3.

As indicated in Table 6.3, the devices with maximum field at the superconducting coils of 8 T are smaller than their 10-T counterparts, and the superconducting devices are smaller than their hybrid counterparts. Contributory factors that lead to these results are described in the following:

1. The higher-field devices have smaller plasma radii. This contributes toward a decrease in radial build; however, it is offset by more subtle changes in the following areas.
2. The higher-field devices require more volt-seconds during burn due to a smaller plasma cross section which tends to enlarge the OH bore.
3. The higher-field devices require thicker inboard shields (due to higher wall loading) and thicker TF coils (if 10 T vs 8 T).
4. Thicker inboard shields and thicker TF coils not only make the device larger but increase the distance from the shaping coils

Table 6.3. Key parameters for TFCX design options

Parameter	S/8 T ^a	S/10 T	H/8 T	H/10 T
Major radius (m)	3.55	3.75	3.60	4.38
Minor radius (m)	1.45	1.07	0.97	0.82
Aspect ratio	2.44	3.51	3.71	5.35
Bucking cylinder inner inner radius (m)	0.74	0.86	1.12	1.62
OH current (at beginning of burn) (%)	-91	-83	0	+42
Volt-seconds required (v-s)	12.6	17.1	18.9	25.5
Plasma current (MA)	10.8	7.7	7.2	5.8
PF system (MAT)	68	79	78	85
TF/CI ^b (MAT)	46/0	81/0	63/24	121/34
ICRH power (MW)	15	24	25	43
Field on-axis (T)	2.62	4.34	4.84	7.10
Beta	0.085	0.059	0.056	0.039
Electron density (10^{14} cm^{-3})	0.52	1.00	1.17	0.39
Energy confinement time (s)	3.75	1.97	1.68	1.14
Neutron wall load ($\text{MW} \cdot \text{m}^{-2}$)	0.3	0.9	1.1	2.1
Fusion power (MW)	110	230	250	490

^aNotation S/8 T refers to "all-superconducting" TF coils/maximum field at superconducting (s/c) coils of 8 T; S/10 T refers to all-superconducting/10 T at the s/c coils; H/8 T refers to superconducting-copper hybrid TF coils/8 T maximum field at the s/c coils; H/10 T refers to hybrid TF coils/10 T at the s/c coils.

^bTF/CI $\equiv$ toroidal field coils/copper insert coils.

to the plasma edge. In the higher-field devices, the plasma edge is closer to the plasma center of current. As a result, shaping is more difficult, requiring larger shaping coils, which limits the OH swing and tends to enlarge the OH bore.

Based on these results, it was decided (in agreement with PPPL) to pursue the 10-T all-superconducting point as TFCX-S and the 8-T superconducting/copper hybrid point as TFCX-H. The characteristics of these designs will form the basis for later comparison with a preferred all-copper TF design.

7. MAGNETIC SYSTEMS

The magnetic systems consist of (1) the TF coils; (2) the PF coils, which include the OH solenoid and equilibrium field (EF) coils external to the TF coil bore; and (3) the cryostat for options involving superconducting coils. The overall arrangement of the magnets is depicted in Fig. 7.1, and the key parameters of the magnets are given in Table 7.1. Each of the systems is briefly described below.

The magnet systems for two superconducting TFCX options are described. In the first option, designated TFCX-S, the required field-on-axis is developed entirely by 16 superconducting TF coils. In the alternative option, designated TFCX-H, the field-on-axis is developed by a combination of 16 superconducting TF coils and 16 copper insert coils located in the plane of and inside the bore of the superconducting coils. The TFCX-H copper insert coils are embedded in the shield modules, and the device may be operated with or without the copper insert coils energized.

7.1 SUPERCONDUCTING TF COILS

In the TFCX-S option, the TF coils produce a field-on-axis of 4.3 T at a major radius of 3.75 m and develop a maximum field of 9.7 T at the TF coil. Four different conductor/coolant concepts have been considered for this application. These concepts are discussed separately. Any of the four will fit within the space envelope allocated to the TF coils and will satisfy the field-on-axis requirement.

In the TFCX-H option, the superconducting coils operate at a peak field level of ~ 8.0 T (rather than ~ 10 T as in TFCX-S) and alone can produce 3.5 T on-axis at a major radius of 3.60 m. When the copper coils are energized, the field-on-axis is increased to 4.8 T, whereas the maximum field at the copper coil is 7.6 T. In both options, the ripple in the toroidal field is limited to 1.5% at the edge of the plasma and 0.1% at the center of the plasma.

The superconducting coils are designed to be cryostable against the combined heat load of resistive heating, nuclear heating, and eddy current heating. The magnet systems are required to withstand magnetic

Fig. 7.1. Overall arrangement of the magnets.

Table 7.1. FED magnetic system parameters (TFCX-S)

Description	At 10 T
Major axis (m)	3.75
Field on-axis (T)	4.3
Number of pulses	1.1×10^5
<u>TF coils</u>	
Number	16
Conductor	NbTi forced flow
Design field at winding (T)	10
Winding bore (m)	6.1×4.2
Maximum permissible radiation dose (rad)	10^9
Ampere-turns per coil (MAT)	5.06
Overall current density (A/cm^2)	1470
Operating current (kA)	28
Stored energy/coil (MJ)	470
Coil mean perimeter (m)	18.8
Bucking post radius (m)	
Outside	1.72
Inside	1.06
<u>PF coils</u>	
Volt-seconds (Wb)	17
Conductor	NbTi
Maximum field at coil (T)	8
Charging time (s)	30
Discharge time (s)	30
Length of central solenoid (m)	6
Winding current density (A/cm^2)	1500
Stored energy in the solenoid (GJ)	1.0

loads arising during normal and abnormal conditions, as well as dead weight and 1-g horizontal and/or vertical seismic loading, while satisfying a variety of access and clearance constraints.

The functional requirements that must be satisfied by the TF coils are included in the following:

- Inboard scrape-off distance requirements
- Shielding sufficient to limit the nuclear heat generation rate and the displacements per atom (dpa) damage in the inboard leg of the TF coil to an acceptable level
- Space occupied by the vacuum vessel
- Gaps between the vacuum vessel and TF coils and between the vacuum vessel and the shield sufficient to preclude contact during credible seismic events
- Space occupied by the TF coil case, influenced by design-allowable working stresses
- TF coil winding space, determined by the ampere-turns needed to satisfy the field-on-axis requirement and by realistically achievable current densities for the conductor concept selected
- Space required by the locking post [The inside diameter is determined by the space requirements of the OH solenoid, which provides the necessary volt-seconds during startup; the radial build (thickness) is determined by the net centering load on the TF coil and the design-allowable stress for the cylinder material(s).]

The conductor and winding are designed to be cryostable. The maximum heat flux that can be dissipated without the loss of superconductivity of the conductor (critical heat flux) is characteristic of the conductor configuration. The coil is designed so that the actual heat flux resulting from the combined effect of normal resistive heating, eddy current heating, and nuclear heating does not exceed the critical heat flux for the conductor.

Conductors cooled with forced-flow helium cannot, however, be characterized as cryostable in the same sense as the conductor cooled with pool-boiling helium. Force-cooled conductors considered in this

report are the cable-in-conduit type. Individual strands in such a conductor can go normal due to localized strand movement. An energy deposition tolerance limit for such conductors is generally determined experimentally.

The critical current density (maximum current density at which the conductor remains superconducting) is degraded when a mechanical strain is imposed upon the conductor. The net strain from all sources is usually limited to about 1.5%-2% for NbTi conductors, and to about 0.5% for Nb₃Sn conductors.

Structural design and analysis of the TF coils and associated components is in accordance with the structural criteria. These criteria consider potential ductile failure due to gross overloading as well as failure due to growth of a postulated structural defect to critical size under the action of a cyclic (pulsed) load. The structural criteria that address ductile failure provide, for normal loading conditions, a margin of at least 1.5 against formation of a fully plastic section in either membrane loading or bending and a margin of at least 3 against rupture. For abnormal loading conditions, the required margins are two-thirds of those for normal loading conditions. The criteria that deal with crack growth considerations provide a margin of at least 3 against critical stress after one design life and a margin of at least 2 after two design lives.

Maximum allowable discharge voltage is limited to 3000 V for the magnets cooled with pool-boiling helium. For force-cooled magnets, this limit can be 10,000 V.

The peak temperature rise of the winding in the event of a quench is limited to 200 K. This limit is primarily governed by the thermal stresses in the hot zone of the winding.

Average current density values for the winding pack and for the overall coil cross section have been adopted. These figures depend upon the conductor concept, the peak field at the winding, the winding operating temperature, and the selected cooling option. The current densities used in studies of the present TF coil design are listed in Table 7.2.

Table 7.2. TF coil current densities

Conductor	Peak field (T)	Temperature (K)	Cooling option*	Current density (A/cm ²)	
				coil	winding
Nbti	8	4.2	PB	1500	2600
NbTi	8	4.2	FF	1500	2600
NbTi	10	1.8	PB	1400	2000
NbTi	10	3.0	FF	1400	2000
Nb ₃ Sn	10	4.2	FF	1400	2000

*PB = Pool boiling; FF = Forced flow

The selection of a TF coil winding insulating material is influenced by the lifetime-integrated radiation dose to which the winding will be exposed, since the mechanical strength of some insulating materials is seriously degraded by nuclear radiation. The maximum achievable current density is influenced by the compressive strength of the insulating material. Glass-reinforced epoxies such as G-10 and G-11 begin to show a marked decrease in strength after about 2×10^8 rad exposure. It is not cost-effective to provide enough shielding to limit the winding fluence to 2×10^8 rad, nor is it cost-effective to base the winding design on the degraded strength of irradiated material. Consequently, use of G-10 or similar insulating materials is not judged feasible for the TF coils if the dose exceeds 2×10^8 rad.

Polyimide insulators have been demonstrated to have superior radiation resistance. In particular, Keramid polyimide, when loaded in compression normal to the plane of the material, has a compressive strength for unirradiated material that is quite competitive with unirradiated G-10CR (~ 120 ksi at 77 K); furthermore, within the error bands of the strength measurements, the compressive strength is unchanged for gamma-ray doses up to 10^{10} rad. For the TFCX design, the maximum fluence on the TF coils is about 6×10^8 rad; consequently, Keramid polyimide is specified as the winding insulator material.

Nuclear radiation induces lattice displacements in the copper stabilizer material. This dpa damage results in an increase in the resistivity of the copper and, therefore, in the amount of resistive heat generation, should a portion of the winding become normal. To ensure that the winding remains cryostable, the combined heat load from resistive heating, nuclear heating, and eddy current heating must be less than the critical heat flux for the selected conductor. The resistivity increase due to dpa damage and the volumetric heat generation rate due to nuclear heating are principal factors in determining the shield thickness in the inboard centerpost region.

A variety of conductor/coolant concepts have been considered for use in TFCX. Each of these concepts can be accommodated in the allotted TF coil space envelope and is capable of meeting the field-on-axis requirement. These conductor/coolant concepts are as follows:

1. Superfluid Cooled NbTi Conductor. A NbTi conductor (three-level Rutherford cable) is cooled in a bath of superfluid helium (He_{II}) at a temperature of 1.8 K. Use of He_{II} coolant extends the peak field capability of NbTi conductors to about 10 T, compared to about 8 T when 4.2 K liquid helium (He_{I}) is used as the coolant.
2. Forced Flow Conductor in Support Plates. A NbTi conductor in a stainless steel conduit is force-cooled by pressurized supercritical helium, which exits the winding at a temperature of about 3.1 K. The conduits are supported in grooved support plates similar to the Westinghouse Large Coil Program (LCP) coil.
3. Forced-Flow Conductor in Epoxy-Bonded Winding Pack. The conductor is similar to the previous case; however, instead of grooved support plates, the cable-in-conduit is cowound with a stainless steel support channel, and the entire winding is epoxy impregnated.
4. Internally/Externally Cooled Conductor in Epoxy-Bonded Winding Pack. Nb_3Sn strands in a conduit are cooled by stagnant supercritical helium inside the conduit. Heat is transferred through the conduit wall to an external 4.2 K bath of helium in grooves cut into the interpancake insulation.

In the TFCX-H option, the peak field at the superconducting TF coil is ~ 8.0 T, which can be achieved using any of the existing LCP conductors. For the present design, the General Dynamics LCP conductor concept is being used for illustrative purposes.

The copper insert coils in TFCX-H are energized apart from the superconducting coils. The machine may be operated with or without the copper coils energized; when the copper coils are energized, they increase the field-on-axis by ~ 1.3 T and account for 27% of the field-on-axis. The copper coils are operated as pulsed devices in order to avoid the excessive resistive power consumption that would result from steady-state energization.

Since copper has about the same shielding capability as steel, use of copper insert coils embedded in the shield offers the advantage of a more compact machine for a given field-on-axis, since the copper serves a dual role. In addition, since the maximum field at the superconducting

coils is less in the hybrid concept than in the all-superconducting concept, the superconducting coils may be designed to a higher current density, helping to make the device more compact. On the other hand, the magnetic forces acting on the copper coils introduce additional complexity from the standpoint of structural support. Since the shield modules are physically separated from the superconducting coils, the centering force developed on the copper coils cannot be transmitted to the bucking post and must be reacted in wedging action. In addition to requiring precise fit-up of the shield modules, this portion of the shield must be fabricated from a high-strength steel in order to avoid an excessive radial build. The out-of-plane (overturning) forces on the copper insert coils also present a structural support problem because of the limited space available inside the bore of the superconducting coils to accommodate intercoil support structure (ISS) for the copper insert coils.

Table 7.3 lists the TF magnet system parameters for the all-superconducting and hybrid options.

During normal operation, there are distributed forces acting along the periphery of the coil, both in and normal to the plane of the coil. The in-plane forces are radially inward and are reacted principally by the bucking post. The out-of-plane forces on the TF coil result in a large overturning moment that tends to tip the TF coils about their horizontal axis of symmetry. This overturning moment is reacted by the ISS, which joins adjacent TF coils at the top and bottom.

To permit removal of torus sectors, there is an "open window" region between adjacent TF coils extending from the horizontal midplane of the TF coils. The upper and lower portions of the TF coils are joined to the ISS modules. Support of the open window region of the TF coils against out-of-plane loads is one of the more difficult TF coil design problems. The need to maintain the open window rules out the use of shear panels or cross-bracing between TF coils in the midspan region. The problem is intensified by the pulsed nature of the out-of-plane loads, which leads to a comparatively low design stress level because of flaw growth and fracture mechanics considerations.

Table 7.3. TFCX TF magnet system parameters

Description	TFCX	TFCX-H	
	S/C coil	S/C coil	Normal coil
Number of coils	16	16	16
Number of pulses	1.1×10^5	1.1×10^5	1.1×10^5
Major radius (m)	3.75	3.60	3.60
Field on-axis (T)			
w/o copper inserts	4.3	3.5	3.5
w/copper inserts			4.8
Field at S/C coil (T)	9.7	7.8	
Field at copper coil (T)			7.6
Winding bore size (M)	$\sim 6 \times 4$	$\sim 6 \times 4$	
Ampere-turns per coil (MAT)			
S/C	5.16	3.94	
Copper			1.50
Mean perimeter, S/C (m)	~ 19	~ 19	
Winding current density (A/cm ²)			
S/C	2000	2600	
Copper (inboard)			1250
Operating current (A)			
S/C	18,100	10,000	
Copper			
Number of turns			
S/C	280		
Copper			
Stored energy per coil (MJ)			
SC	470	210	
Copper			100
Weight per coil (10 ³ kg)			
SC	44.4		
Copper			

The TF coils are contained within a common vacuum vessel. The vacuum vessel is designed to enclose the open window region of each TF coil in a manner resembling the fingers of a glove. In this way, the open access region between TF coils is preserved.

7.2 PF COIL SYSTEM

The PF coil system consists of OH solenoid that provides energy to the plasma during the burn phase and a set of EF coils to maintain plasma equilibrium and produce a plasma triangularity of 0.3. All PF coils are located outside the bore of the TF coils and are superconducting. A constraint adopted in determining the size and location of the PF coils is to limit the field in each of these coils to ≤ 8 T in order that NbTi conductors may be used.

The PF system performs two main functions: (1) to shape and equilibrate the plasma and (2) to inductively heat the plasma. The design objective is to satisfy these requirements at minimum PF system cost. Individual coil locations should be such that any coil could be easily maintained and replaced if necessary. The PF coil also must not inhibit maintenance or replacement of any other subsystem.

The functional requirements that must be satisfied by the PF coils are (1) the PF system must provide 0.3 triangularity with 1.6 elongation of the plasma, (2) the peak field at any coil must not exceed 8 T, (3) adequate volt-seconds must be provided to allow for a 300-s burn, and (4) the PF configuration must be consistent with the TF and other structures.

The PF coils must be designed to be stable for operational heat loads. They must be structurally adequate to carry magnetic loads, dead weight, and 1-g seismic loads. The coils must be capable of being rapidly discharged from the fully charged state without damage and must survive all credible abnormal conditions without damage.

All of the PF coils are categorized as semi-permanent structures (i.e., they are to be designed for the life of the device and are not intended to be replaced or repaired). Nevertheless, the design must be

such that coils can be replaced, if necessary. The design criteria for the PF coils are the same as those for the TF coils discussed above.

Key features of the solenoid design are a double-pancake winding layout, modular design with modules powered by separate leads, a glass epoxy helium vessel to avoid induced currents, leads located in the central bore to minimize space required, and a solenoid supported off the TF ISS. The conductor for the solenoid is a 50-kA NbTi, copper-stabilized, pool-boiling, liquid-helium-cooled, flat cable. The conductor consists of 36 subcables around a central core made up of two strips of Nitronic 40. The use of two strips makes the core more flexible than a single-thickness strip would be, facilitates winding, and reduces eddy-current losses when an insulating material sheet is sandwiched between the two strips. Each subcable consists of six insulated superconducting strands around a stainless steel strand. The stainless steel strand increases the tensile strength of the subcable. The turn-to-turn insulation consists of G-10 spacers cowound with the conductor.

Key design features of these superconducting ring coils are (1) a 50-kA cable conductor; (2) pancake winding; (3) a stainless steel, ring-stiffened case to provide structural support and to contain helium; (4) sliding pedestal supports to transmit dead weight, seismic loads, and the resultant out-of-plane magnetic load to the ISS; (5) a coil case designed to equilibrate magnetic bursting load and magnetic bending loads; (6) dielectric breaks in the case to reduce induced poloidal currents; and (7) joints sealed by welded omega seals or double O-rings. The conductor being used for the superconducting ring coils is similar to that chosen for the solenoid, that is, a 50-kA bath-cooled (by liquid helium at atmospheric pressure), cabled NbTi conductor.

The arrangement of the OH solenoid and the superconducting EF ring coils in the TFCX-S design is shown in Fig. 7.2.

7.5 CRYOSTAT

A common vacuum cryostat contains all of the superconducting coils. The cryostat has separate individual enclosures for the outboard legs of the TF coils. This approach maintains the good access between the TF

Fig. 7.2. Arrangement of OH solenoid and superconducting EF ring coils in the TFCX-S design.

coils and requires no penetration of the cryostat boundary for torus access. This approach also separates the warm and cold components of the configuration. The key elements of the cryostat vessel include a cold wall inside the vacuum vessel following the contours of the vacuum vessel and individual helium vessels for each superconducting coil.

The vacuum vessel is a shell-of-revolution around the entire magnetic system except for a portion of the vertical legs of the TF coils at their outer extremity, where openings are left to permit access to the torus. (Glovelike extensions of the vessel enclose these vertical legs.) The material is type 304 stainless steel. Preliminary calculations show that insulating breaks will not be needed.

8. NUCLEAR SYSTEMS

The nuclear systems consist of those components that make up the toroidal plasma chamber. For TFCX-S, those components include the torus support platform and pedestal, common vacuum boundary cryostat, removable shield module, pumped limiter, shield post, and first wall system. The general configuration of these components is shown in Fig. 8.1.

For TFCX-H those components include the torus support platform and pedestal, common vacuum boundary cryostat, removable shield module, copper TF coil/shield module, shield wedging post, connecting intercoil structure, fixed window shield modules, pumped limiter, and first wall system. Figure 8.2 shows the general arrangement of TFCX-H.

8.1 TORUS AND SHIELD DESIGN

Assembly of the shielding components within the common vacuum boundary cryostat on the torus support platform makes up the toroidal plasma chamber. The resulting torus is built within the bore of the 16 superconducting TF coils. The cryostat has window enclosures running from the toroidal chamber radially outward between the outboard TF coil legs. All the shielding components are installed through flanged openings in the 16 cryostat windows. This approach separates the hot and cold components and allows access to the plasma chamber without penetrating the separate cryostat vacuum vessel boundary for coils located internal to the reactor vessel.

The shielding components within the plasma chamber attenuate nuclear radiation, convert neutron kinetic energy into heat, provide for removal of this heat, and support the first wall and limiter components. Other system components such as rf heating, fueling, testing, diagnostics, instrumentation, and control are also installed within the shielding.

The inboard shield thickness for TFCX-S and TFCX-H is 0.58 m and 0.59 m, respectively. The shielding materials are type 316 stainless steel (80 v/o) and water (20 v/o), with the innermost 0.3 m made of B_4C

Fig. 8.1. TFCX-S general configuration.

Fig. 8.2. TFCX-H general arrangement.

contained in a thin wall container. The stainless steel is in plate form, with machined coolant passes, encased in a pressure-containing shell.

The outboard shield thickness for TFCX-S and TFCX-H is 2.2 m and 2.6 m, respectively, in the window region. The shield materials are type 316 stainless steel (60 v/o) and water (40 v/o) for the first 0.70 m from the plasma. The stainless steel is in roughly spherical-shaped ball form (1.27 cm), procured from ball-bearing manufacturers. The remaining outboard shield material is water (95 v/o) contained in structural boxes. The shielding is contained in a compartmentalized stainless steel pressure vessel with layers of B_4C (0.03 m) and lead (0.05 m) on the outboard face, respectively.

The outboard shield thickness in the plane of the TF coil is 0.85 m for both the TFCX-S and TFCX-H configurations. The shield materials for TFCX-S are 316 stainless steel (60 v/o) in the ball form and water (40 v/o), with layers of B_4C (0.03 m) and lead (0.05 m) on the outboard face. The materials for TFCX-H will be the same, except that the percentages and form will be determined by structural requirements of the copper coil insert.

The shielding thicknesses were determined by the requirements to (1) limit the nuclear heating rate in the TF coil winding (inboard leg) to 1 mW/cm^3 and (2) limit the shutdown dose rate after 24 h to 2.5 mrem/h .

The structural component of the bulk shield is a welded structural box made of Nitronic 33. The bulk shield steel plates (Nitronic 33) in the inner shield region are bolted to each other and to the structural box and are coated with a dielectric (Al_2O_3) to minimize eddy currents induced by plasma disruptions. The coolant water is channeled poloidally through machined grooves in the steel plates. There is no effort to seal the coolant pressure in the grooves. Rather, the box structure, stiffened by the bolted plates, carries the coolant pressures. The thickness of the plates and the spacing and size of the coolant grooves enable the desired distribution of steel and water to be achieved within 1%.

In the outer blanket region, the structure is a welded box that is used as a container for the stainless steel balls that fill the structural box in the first 0.70 m of the outer shield region.

The torus support system consists of 16 vertical columns, a toroidal support platform, and lateral support panels. The torus shield sectors are supported by the support platform and vertical columns, one of which is located under each shield sector. The lateral support panels provide a base for installation and removal of the shield sectors. All support system components are constructed of type 304 stainless steel and operate at room temperature. The vertical support column surround the pumping duct channels that pump out the plasma chamber and provide impurity control pumping during the burn.

Sector removal/replacement is accomplished by moving the sector radially outward on integrally mounted roller-dolly assemblies. Once clear of the device, the sector is transported by an overhead crane.

8.2 FIRST WALL SYSTEM

The interior of the plasma chamber is designed to protect the basic torus from heat loads and energetic particles generated by the plasma and to protect the plasma from deleterious impurities generated by interaction of the plasma with the interior surfaces. Of special concern are the inboard, top, and bottom surfaces that are subject to very high, but poorly defined, heat loads during disruption.

The TFCX first wall system includes all components which directly face the plasma. It consists of the pumped limiter on the bottom of the plasma cavity, passively cooled armor tiles on the inboard wall, and actively cooled sandwich panels on the outboard wall. The panels are made of 316 stainless steel. The design concept is shown in Fig. 8-3. The pumped limiter is technically part of the first wall system, but due to its special function of impurity control, it is discussed separately.

The primary purpose of the first wall system is to accommodate the nominal heat loads during startup and burn and the anticipated disruption energy without replacement for the life of the device. The thickness

Fig. 8.3. TFCX first wall design concepts.

for both the armor tiles and the stainless steel panels was assumed to be 5 cm in the radial build. Specific thermodynamic analyses using startup, burn, and anticipated disruption data for the TFCX must be performed in the future to confirm that 5-cm thickness is the appropriate thickness. Candidate materials under consideration for the armor tiles presently include BeO, Be, B, B₄C, C and possibly SiC.

To accommodate the surface heat loads during normal operation, water-cooled first-wall panels are installed on the outboard shield facets. These panels are fabricated of type 316 stainless steel. The panels are a multipass design, with the coolant flowing toroidally back and forth through the panels in a serpentine pattern.

The panels are installed from the sides of the shield sectors. Removal of the sector is required in order to replace a panel. Panel replacement should be infrequent, however, since the panel life is expected to exceed the life of the device.

8.3 IMPURITY CONTROL SYSTEM

The TFCX employs a mechanical pumped limiter for particle and impurity control. The purpose of the limiter is to (1) establish the plasma edge, (2) provide a mechanism to remove helium ash and hydrogen particles from the plasma, and (3) help protect first-wall components from large particle and energy fluxes.

The limiter is located in the bottom of the plasma chamber mounted in a cantilever position from the outboard shield, as shown in Fig. 8.4. The limiter is divided into 32 segments, with two segments installed in each removable shield module through an access penetration. A flanged vacuum seal on each penetration at the outboard face of the shield module allows removal and installation of limiter segments without removing the shield module. The limiter segments come together in a radial, spokelike fashion forming a toroidally continuous limiter surface inside the plasma chamber, as shown in Fig. 8.5. The inboard tip of the limiter surface is 32 cm from the plasma tangent line.

Fig. 8.4. Plan view of installed limiter segments.

Fig. 3-5. TFCX limiter location.

Each limiter segment has a reusable core structure and a replaceable heat ablative protective sleeve. The core structure is made of Nitronic 33 stainless steel, and the protective sleeve consists of a sandwich panel, copper substrate with brazed armor tiles. Candidate armor tile materials are as previously indicated. Both the core structure and the copper substrate are actively cooled. Limiter segments are electrically connected by pressurized bellows that activate copper contacts along one edge of each segment.

9. PLASMA HEATING SYSTEMS

The plasma heating systems consist of those systems needed to perform plasma initiation, current ramp-up, and plasma bulk heating. The initiation phase includes the ionization or breakdown of the gas and heating beyond the point where radiation losses dominate the energy balance. During the current ramp-up phase, rf energy is planned to increase the current to near the final values. In the final phase, auxiliary heating provides the energy to reach and maintain the plasma burn conditions.

9.1 PLASMA INITIATION AND STARTUP

TFCX uses an rf system to provide initial heating. This system overcomes the need for a high-voltage plasma initiation and reduces the volt-seconds required from the PF system. About 1 MW of ECRH is applied when the gas is puffed into the torus. This rf energy is sufficient to ionize and heat the electrons in the absence of a toroidal plasma current. The electron cyclotron wave is launched from the high-field side (HFS) of the torus using the extraordinary mode of propagation. Ionization occurs in a relatively small vertical band around the major radius, whose radial position is determined by the frequency of the rf system and the magnitude of the toroidal field. A few hundred milliseconds after the ECRH is applied, the temperature at this vertical band of plasma exceeds the temperature at which radiation loss dominates. Plasma current is started after the electron temperature reaches ~ 100 eV.

The ECRH system configuration consists of a high-power gyrotron oscillator that supplies a mode-transducing launcher through an overmoded circular waveguide. The ECRH system is a single-frequency system (105 GHz) that provides preheating of the electrons to a temperature of ~ 50 to 100 eV, which is above the radiation barrier. The requirements of the ECRH system are given in Table 9.1 for both the TFCX-S and TFCX-H concepts.

Table 9.1. ECRH preheating requirements
for TFCX-S and TFCX-H

	TFCX-S	TFCX-H
Type	ECRH	ECRH
Species	D	D
Pulse length capability (s)	3	3
Power (MW)	1	1
Frequency (GHz)	105	115
Field at startup (T)	3.75	4.11
Field on-axis (T)	4.34	4.84
Mode	χ^a	χ^a
Launcher position	HFS ^{<i>b</i>}	HFS ^{<i>b</i>}
Injection angles (degrees)	85	85

^{*a*} χ — the extraordinary mode.

^{*b*}HFS — the high-field side.

9.2 HEATING AND CURRENT DRIVE

It is proposed to use rf systems to ramp up the current subsequent to the ionization of the plasma. It is planned to ramp up the current slowly (~ 30 s) to its full value using about 13 MW of LHRH power provided through Brambilla grills. This should permit driving the current up to about 7 MA. Table 9.2 summarizes the requirements for current drive for both TFCX-S and TFCX-H.

The bulk heating of the plasma is planned to occur following the current ramp-up. During this period, the bulk heating is provided by about 25 MW of ICRH from a poloidal stack of four ridge-loaded waveguides. The ICRH system has some tuning capability to allow operation at lower magnetic fields and/or with different fuel species during the check out phase. This system will raise the plasma temperature and density to operating conditions in about 10 s. The requirements for the ICRH bulk heating system are given in Table 9.3 for TFCX-S and TFCX-H.

During the burn portion of the plasma cycle, current is supplied by the OH transformer. If operation is conducted in nonignited conditions (i.e., the device is operated in a driven mode), bulk heating of the plasma is provided by the LHRH and ICRH systems. Burn pulse lengths of ~ 300 s have been specified for both TFCX-S and TFCX-H.

The requirements for the plasma heating systems are traceable to plasma performance through scaling laws that are based on analyses/modeling and experimental results. At present, the understanding of current drive and heating efficiency of various modes is growing rapidly. Because of this, the requirements are continuously evolving. If a single system could be used to perform all functions at the required performance level, this would be ideal; however, at present there is little or no substantiation that such a concept will evolve from present-day experiments. Startup experiments with rf are under way at PPPL using LHRH. If successful, consideration could be given to eliminating the ECRH preheating system. Current-drive experiments using ICRH at PPPL and bulk heating experiments using LHRH at PPPL and Massachusetts Institute of Technology (MIT) are either planned or under way. The results of these could allow elimination of one system and lower the

Table 9.2. Current-drive requirements
for TFCX-S and TFCX-H

	TFCX-S	TFCX-H
Type		
Initiation and startup	LHRH	LHRH
Bulk heating	LHRH	LHRH
Burn	OH	OH
Species	H, D, D-T	H, D, D-T
Pulse length (s)		
H or D	340	340
DT	40	40
Power (MW)		
H or D	14	12
D-T	14	12
Control	10:1	10:1
Frequency (GHz)	2.73	2.73
Field on axis	4.34	4.84
Density (cm ⁻³)		
High	1.8×10^{14}	1.8×10^{14}
Low	1.3×10^{13}	1.3×10^{13}
Mode	0	0
Launcher position	Midplane LFS	Midplane LFS ^b
Spectrum control (n ₁₁)		
Low density	1.4 ± 0.1	1.4 ± 0.1
High density	2.1 ± 0.1	2.1 ± 0.1

^a0 - the ordinary mode.^bLFS - the low-field side.

Table 9.3. Bulk heating requirements

	TFCX-S	TFCX-H
Type	ICRH	ICRH
Species		
Startup	H, D&H minority	H, D&H minority
Bulk heating	H, D, D-T	H, D, D-T
Burn	H, D	H, D
Harmonic		
H	1st	1st
D	2nd	2nd
D-T	2nd D ⁺	2nd D ⁺
Frequency (GHz)	0.0651	0.0726
Field on-axis (T)	4.34	4.84
Mode	χ^a	χ^a
Launcher Position	Midplane LFS ^b	Midplane LFS ^b
Power (MW)		
H, D	36	42
D-T	23	25
Control	20:1	20:1
Pulse length (s)		
H, D	340	340
D-T	40	40

^a χ - the extraordinary mode.

^bLFS - the low-field side.

cost and complexity considerably. The use of the three rf systems combined with the OH transformer provides a very flexible system and allows a wide variety of experiments and scenarios to be considered, as were briefly described in Sect. 5.

Figure 9.1, 9.2, and 9.3 illustrate the basic configurations and features currently envisioned for the ECRH, LHRH, and ICRH systems, respectively.

Fig. 9-1. TCEX ECRH system.

Fig. 9.2. LHRH launcher interface with torus.

Fig. 9.3. ICRH launcher interface.

5.5 SHUTDOWN

During shutdown, the fusion burn is quenched before the current is quenched. Under normal conditions, the fusion quench is initiated by termination of fueling while maintaining or possibly enhancing plasma exhaust through the pump limiter. Transport estimates have suggested a time scale of ~ 2 s is required to return to the ohmic state. A plasma density of $\sim 2 \times 10^{13} \text{ cm}^{-3}$ is assumed at the end of the fusion quench. Conceptually, the plasma current shutdown is the reversal of the plasma current ramp-up. As the plasma radius decreases, the value of q_ψ at the plasma edge should be maintained or increased as much as possible during this phase to avoid disruptive termination of large plasma current.

5.6 PUMPDOWN

The neutral density in the chamber at the end of the current quench is expected to be $\sim 3 \times 10^{-4}$ torr. It is assumed that the residual gas is relatively clean, with only negligible amounts of helium and low-Z impurities. Pumpdown to 3×10^{-6} torr in ≤ 30 s is assumed during this phase.

10. ELECTRICAL SYSTEMS

The TFCX electrical systems consist of power handling and conversion; energy storage; and plasma diagnostics, instrumentation, data acquisition, and control.

10.1 POWER HANDLING AND CONVERSION

The power handling and conversion system consists of the ac power system and the power supplies and protection system for the TF and PF coils. The ac power includes the 60-Hz input from the utility company, the associated substations, and an intercomplex distribution system. Since there has been no site selection at this time, the specific utility line has not been included in the studies. It is understood that there may be some load leveling required, but operation in general will not be curtailed by the load requirements for the superconducting coil options. Two power lines will be required for sites that are not located near major power lines or generating capacity. One line will provide the pulsed power and the other steady-state power and general facilities. In addition, standby diesel generators and uninterruptible power supplies are needed for backup power for essential loads. The major ac power loads for TFCX-S and TFCX-H are given in Table 10.1.

The steady-state or longer duration loads are listed in the upper part of the table. Pulsed loads that do not exist during the burn phase are listed in the lower part of the table. The pulsed loads listed in the lower part of the table are five to ten times higher than the continuous loads. The ECRH load is small compared to the others. Both LHRF and ICRH power loads exist for the entire 30-s startup. If these loads can be ramped on or applied in steps, they can be connected directly to the utility line. The large pulsed PF coil power supply load is ramped up on over a 30-s period. Therefore, the PF coil load can be connected directly to the utility substation. For the same reason, the TFCX-H resistive copper (RC) TF coil power supply load can be connected directly to the utility substation.

Table 10.1. Major Ac power system loads

Description	TFCX-S (MW)	TFCX-H (MW)
<i>Steady power</i>		
Facility power	15	15
Power to SC TF PS	2.4	2.4
Power to RC TF PS		50.0
Power to PF coil burn PS	2.4	2.3
Power to cryogen system	18.9	9.2
Power to cooland system	6.4	7.6
Power to fuel processing system	5.1	5.6
Power to vacuum system	2.7	2.1
Total	53	94
<i>Pulsed power</i>		
Power to LHRH RS (40-s) (40 to 340 s)	39	33
Power to ICRH PS (40-s)	45	68
Power to ECRH (3-s)	5	5
Power to PF coil pulsed PS	190	220
Counterpulse PS		50
Power to RC TF coil PS		20
Total	279	396

Based on the above considerations and the number of electrical loads, a pulsed substation is required that can handle about a 480-MW peak load for TFCX-H and 325 MW for TFCX-S. All the steady-state loads that do not require normal voltage regulation can also be connected to this substation. All other loads must be connected to the facility substation, which provides better voltage regulation. Major essential power loads (listed in Table 10.2) require backup power for operation when the utility power fails. The atmospheric detritiation system is much larger than all other essential power loads. About 40 kW of uninterruptible power is needed for safety-related instrumentation and control.

10.2 ENERGY STORAGE

The energy storage for the superconducting options will be essentially zero. The long rise time and the long pulse essentially eliminate the advantage or need for on-site energy storage. There may possibly be a need for transient control power that would result in a few megajoules of energy storage, but this would be much less than now used in Doublet III or TFTR.

10.3 PLASMA DIAGNOSTICS, INFORMATION, AND CONTROL SYSTEMS

The diagnostic requirements will include the same diagnostics as on present operating tokamaks, but there will also be the need for advancements in features such as current density measurements and plasma shape. The neutron and radiation environment will be greatly increased over present devices, as is generally recognized. This will require extensive remote handling and/or shielding. In addition, the long-pulse periods will require development of steady-state data output rather than one data point per pulse. The instrumentation for measuring the plasma loop current for a 300-s steady-state flat-top will require redevelopment of the existing technology.

Table 10.2. Essential power requirements

Load description	Power (kW)
Emergency tritium cleanup	3000
Fuel reprocessing	500
Essential HVAC	200
Uninterruptible power	20
Data system	50
Instrumentation and control	50
Essential lights	50

11. AUXILIARY SYSTEMS

The auxiliary systems are defined as those supporting systems that, though not physically part of the device, are closely integrated with it in terms of systems operation. The fueling system, vacuum pumping system, tritium systems, and remote maintenance equipment comprise the auxiliary systems and are briefly described below.

11.1 FUELING SYSTEM

The fueling system consists of gas puffers and pellet injectors. Prior to startup, the fuel gas puffers are used to fill the torus with deuterium to the desired pressure. Gas puffing continues until a plasma of ~ 1 keV is established. Frozen deuterium and tritium pellets are then injected to control the plasma density.

After the startup is completed, fuel pellets are injected to control the average plasma density and the gas injectors may be used to control the background gas pressure.

During early phases of operation, when the hydrogen plasma density and temperature are an order of magnitude lower than design conditions, the gas injectors may provide for all the fueling needs. Fueling in later phases of operation with higher plasma density and temperature requires frozen pellet injection for deeper penetration into the plasma. To accommodate all phases of operation, the injectors should have the capability of injecting all three hydrogen isotopes: protium, deuterium, and tritium. Different pellet formers may have to be substituted when changing from hydrogen to deuterium-tritium operation and vice versa. Alternatively, three independent pellet formers may be employed.

Although large pellets are desirable for deeper penetration into a dense plasma, the pellet mass is limited to about 10% of the plasma mass to prevent large perturbations in plasma density and temperature. This consideration and the particle confinement time determines the pellet size and pellet rate at the design point. The pellet size is about 3 mm and the burn pellet rate is about 6 s^{-1} , which needs to be doubled

to allow for uncertainties in confinement time and the additional acceleration needs during startup. The pellet injector and gas injector requirements data are summarized in Tables 11.1 and 11.2, respectively. The pellet velocity is limited to about $2 \text{ km} \cdot \text{s}^{-1}$ based on target goals of the pellet injector development program. A higher velocity would be desirable when operating near design point conditions. At these conditions, a 3-mm-dia pellet injected at $2 \text{ km} \cdot \text{s}^{-1}$ will penetrate the plasma about 25% of the minor radius.

To reduce potential tritium hazards, it is desirable to minimize the pellet injector tritium inventory. Based on a batch-type pellet former, the tritium inventory necessary for a 300-s burn is about 6 g. If repeated 300-s burn cycles are desired, the inventory would be about 10 g, to allow for freezing one batch while injecting the other. Auger-type pellet former-injectors could conceivably reduce the inventory to less than 6 g for continuous operation.

The most effective location for injecting gas is on the underside of the plasma in the vicinity of the limiter blade. The cost of gas injectors is more than an order of magnitude lower than pellet injectors. For this reason, two independent gas injector systems can be justified, each having the capability of injecting at eight locations. Gas injection mass flow rate should be at least ten times the pellet injection rate to accommodate the uncertainty in gas recycle physics of the plasma and to provide for adequate response time in backfilling the torus and controlling the background pressure. The gas injector system should have the capability of injecting any of the hydrogen isotopes or mixtures thereof. The gas flow may be pulse modulated or employ continuous modulation.

11.2 VACUUM PUMPING

The vacuum pumping system for TFCX is designed to pump down the torus vacuum vessel initially and between burns and to remove the gas load from the pumped limiter during the burn period. A detailed study of impurity removal during the burn, which accounts for the interaction of the plasma and neutral gas particles, remains to be done.

Table 11.1. Pellet injector requirements for TFCX

Parameter	TFCX-S and TFCX-H
Number of dual-channel pellet injectors	1
Pellet size (mg)	
Tritium	6.5
Deuterium	4.3
Pellet diameter (mm)	3
Pellet composition	
Hydrogen phase	>99% H ₂
D-T phase	
Tritium pellet	>99% T ₂
Deuterium pellet	>90% T ₂
Mixed pellet	Recovery mixture ^a
Injector distance to plasma (m)	5
Magnetic field in vicinity of pellet injector (T)	TBD
Pellet velocity, max (m·s ⁻¹)	≥2000
Pellet rate/channel, max (s ⁻¹)	6
Maximum injection period (s)	350
Low tritium leakage	(Double confinement)
Low tritium inventory (g)	~10
Availability/8 h	>0.95

^aMixture of fuel gas species exhausted from the torus with impurities removed.

Table 11.2. Gas injector requirements

Parameter	TFCX-S and TFCX-H
Number of deuterium gas injection systems	2
Number of gas injection nozzles	8
Maximum gas flow rate ($\text{g}\cdot\text{s}^{-1}$)	0.3
Maximum valve pulse length (ms)	150
Maximum valve pulse rate (s^{-1})	5
Gas composition	All hydrogen isotopes and mixtures thereof
Availability/8 h	>0.95

Since the vacuum pumping requirements are not fully specified at this time, final selection of the high-vacuum pumps has not been made. The two primary candidates are (1) turbomolecular pumps and (2) compound cryocondensing/cryosorption pumps. An alternative might be a combination of the two. The following description assumes the use of turbomolecular pumps.

Particles enter the toroidally continuous, 0.1-m-high pumping slot underneath the limiter blade. The length of the slot is about 0.5 m, corresponding to the position where plasma particles passing behind the limiter strike the backside of the limiter and are neutralized. The toroidally continuous slot is joined to 16 passages, one in each torus sector. Each of these passages is 0.3 m high, 0.9 m wide, and 1.35 m long. An 0.5-m-diam vacuum duct with two 90° elbows connects each of these passages to a plate valve and $5 \text{ m}^3 \cdot \text{s}^{-1}$ turbomolecular pump. There are eight valves and pumps arranged around the torus; therefore, two of the 0.5-m-diam ducts feed into each of the valve and pump assemblies.

The design guidelines assumed for the TFCX vacuum pumping system performance are listed in Table 11.3. An initial pumpdown pressure of less than 10^{-7} torr was set as a guideline to ensure that impurity levels will not affect plasma performance. The postshot pressure of 2.8×10^{-3} torr was calculated by assuming that the neutral gas density in the plasma chamber after the burn is the same as the plasma density during the burn. The prefill or preshot pressure was assumed to be 10^{-5} torr, which ensures that the number of impurity molecules remaining in the chamber after pumpdown will be less than 1% of the number of D-T full atoms. A pumpdown time of 30 s between consecutive burns was used.

The vacuum pumping system consists of 16 identical vacuum ducts, one in each of the 16 torus sectors, as shown in Fig. 11.1. The 16 ducts merge into one below the limiter, diverging to pass between TF coils on their way to the vacuum pumps. At the end of each pair of ducts is a turbomolecular pump, backed by a scroll pump.

The exhaust from all of the first-stage scroll pumps is combined and passed to a second-stage scroll pump that compresses the gases to

Table 11.3. Guidelines for TFCX vacuum pumping system performance

Condition	Guideline	Basis
Initial plasma chamber pumpdown pressure (torr)	$\leq 10^{-7}$	Maintain low impurity level
Postshot pressure (torr)	2.8×10^{-3}	Calculated
Preshot pressure (torr)	10^{-5}	Impurity content of <1% in plasma chamber after fueling
Pumpdown time between burns (s)	30	Reasonable dwell time between burns
Conducting ratio, γ (%)	20	Ensure adequate impurity removal during the burn

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Fig. 11-1. Vacuum pumping system.

approximately atmospheric pressure. The major design parameters of the vacuum pumps are summarized in Table 11.4.

The high-vacuum turbomolecular pumps are required to have a speed of $5 \text{ m}^3 \cdot \text{s}^{-1}$ for helium and hydrogen isotopes. They will have to be designed for tritium service; all seals will have to be metallic; and provisions will have to be made for remote oil change, maintenance, repair, and replacement. A further consideration is the possibility of developing a $5\text{-m}^3 \cdot \text{s}^{-1}$ turbomolecular pump with magnetic or gas bearings to avoid the hazards associated with tritium-contaminated oil in conventional oil-bearing pumps.

Scroll pumps back the turbomolecular pumps and are used for rough pumping of the plasma chamber. These are sealed pumps that have no bearing lubricant in contact with the tritium stream. Two stages of scroll pumps are required to reach the pressure of 10^{-4} to 10^{-3} torr needed for effective pumping by the turbomolecular pumps to pressures below 10^{-7} torr. A first-stage scroll pump backs each turbomolecular pump, exhausting at a pressure of about 25 torr. The exhaust from all eight first-stage pumps is combined and ducted to a single, small, second-stage scroll pump that exhausts at about atmospheric pressure.

The effective pump speed of the total vacuum pumping system as currently designed is $18.5 \text{ m}^3 \cdot \text{s}^{-1}$. The major limitations of the total system are the conductance of the 0.5-m-diam duct and the pump speed. A duct diameter greater than 0.5 m cannot be used for the reference TFCX major radius, duct routing scheme, and TF coil configuration. The duct size can increase significantly if the major radius is increased. If the total pump speed is increased from $40 \text{ m}^3 \cdot \text{s}^{-1}$ [8-5 $\text{m}^3 \cdot \text{s}^{-1}$ (e.g., 32-5 $\text{m}^3 \cdot \text{s}^{-1}$ pumps or 8-20 $\text{m}^3 \cdot \text{s}^{-1}$ pumps), the effective pump speed of the total system is increased to $33 \text{ m}^3 \cdot \text{s}^{-1}$. If, in addition to the increased pump speed, a configuration that allowed 1.0-m-diam vacuum ducts was developed, the effective pump speed would be $60 \text{ m}^3 \cdot \text{s}^{-1}$. The actual design parameters to be used for TFCX remain to be determined, but the vacuum pumping requirements may not be easily satisfied.

Evaluation of the plasma exhaust pumping indicates that the conductance ratio for the reference design is 5%, which falls short of the 20% guideline. By simply replacing the $5 \text{ m}^3 \cdot \text{s}^{-1}$ pumps with $20 \text{ m}^3 \cdot \text{s}^{-1}$

Table 11.4. Vacuum pump parameters

High-vacuum pumps

Pump type	Turbomolecular
Pump speed for H_2 and H_e	$5 \text{ m}^3 \cdot \text{s}^{-1}$
Number of pump ducts	16
Number of pumps per duct	1 pump per 2 ducts

Backing/roughing pumps

Pump type	Scroll
Number of stages	2
Number of first-stage pumps	8
Number of second-stage pumps	1
Speed of first-stage pump	$35 \text{ l} \cdot \text{s}^{-1}$
Inlet pressure	10^{-4} to 10^{-3} torr
Speed of second-stage pump	$\sim 2 \text{ l} \cdot \text{s}^{-1}$
Inlet pressure	~ 25 torr
Discharge pressure	~ 1 atm

pumps, the conductance ratio is increased to 9%. Although the diameter of the vacuum pumping ducts cannot be increased beyond 0.5 m in some space-limited regions (e.g., between TF coils), significant improvements may be possible by increasing the duct size in other regions. A major vacuum system configuration change may be possible in the region immediately outboard of 4 of the 16 removable sector modules, and this would also improve the conductance ratio. Future effort is needed to determine which, if any, design changes are required to establish adequate impurity removal efficiency.

11.3 TRITIUM SYSTEMS

The tritium systems serve two primary functions: (1) to provide new and/or recycled tritium and deuterium fuel for plasma operations and (2) to limit worker radiation exposure and to minimize tritium losses to the environment.

The TFCX class of fusion device will have a nominal 2×10^5 s of tritium operation over a lifetime of five to ten years. The power level is expected to range from 250 to 400 MW with an assumed fractional burn of 2%. This device uses a pellet fueler and turbomolecular pumps. Tritium neutral beams are not used. Their use would increase the tritium needed by a factor of 3.

To assess the tritium supply, the tritium systems needed, and the environmental impact of this class of device, two burn times (20-s pulses and 300-s pulses) and different processing modes (on-site and off-site, continuous and batch) were considered.

A number of trade studies were performed to determine the effect of different factors in designing tritium systems for a TFCX class of device; (1) the effects of plasma characteristics on the amount of tritium needed; (2) an assessment of the hazard associated with transporting tritium; (3) tritium needs and the location within the plant for different processing scenarios; and (4) a summary of the advantages and disadvantages of different processing scenarios.

A reference case was developed that fulfills the requirements desired for plasma physics experiments. A complete fuel cycle is

provided. This includes not only the components needed to process and store the fuel (fuel cleanup unit, isotope separation unit, storage beds, receiving glovebox) but also the units required for secondary containment (glovebox detritiation system), waste processing (tritium waste treatment - gases; tritiated water recovery unit - water and tritiated waste processing - solids), tritium analysis (required to detect problems and their solutions in the fuel cycle and to maintain accountability records), tritium monitoring (safety considerations), and process control with associated data acquisition system. Figure 11.2 shows the flow paths between the tritium systems.

The units are located in a building separate from the reactor that is leaktight and earthquake-proof. Earthquake valves are located on lines and between units to minimize any potential releases. An atmospheric processing system is in the tritium processing building to handle all tritium releases in the area. The floor plan has been arranged so as to have access to all units during maintenance and to minimize the length of connecting lines.

11.4 REMOTE MAINTENANCE EQUIPMENT

Remote maintenance procedures are required whenever it is necessary to remove or gain access to components comprising or located inside the reactor radiation shield system. This includes removal of shield sectors and ancillary equipment shielding and access to components penetrating the shield envelope. In addition, the capability to maintain equipment outside the shield envelope by remote means has to be provided in order to accomplish at least some of the more straightforward operations during radiation cooldown and, if necessary, to carry out whatever operations are required while part of the shielding system is removed.

Two classes of remote maintenance equipment are required: general purpose and special purpose. Special purpose equipment is required for handling and transporting components, for inspecting areas of concern, for detecting faults, for cutting and welding nonstandard joints, and for making in situ repairs. General purpose devices such as manipulators

Fig. 11.2. Tritium flow paths for TFCX tritium processing units.

and cranes, are required for operations involving tools and for handling components once they have been removed from the reactor. In addition, the general purpose equipment must have sufficient flexibility to perform unplanned maintenance operations. Although some limited design has been carried out on specific items of remote maintenance equipment, substantial additional work is required to better define the characteristics of the remote maintenance equipment and to establish a reliable cost estimate. The specific requirements for TFCX have been assessed compared to the requirements established for the FED and the remote maintenance equipment needs have been defined particular to satisfying the mission requirements of the TFCX device.

At present, facilities data are displayed locally and archived on tapes that may be recalled at the local control area. With the advent of a more unified fusion development program, the data can be made available to facilities at remote locations. The operational data can be provided to several institutions or laboratories as they are generated.

The control of the facility and the operation of the device will utilize the state-of-the-art digital control system. The TFCX instrumentation and control system will be based on the inclusion of dedicated local control and protection functions in each subsystem and between subsystems, where required. Integrated facility monitoring and operator interface functions will be provided by a central control facility, which will interface with and overlay subsystem controls. Diagnostic data acquisition will be accomplished by dedicated, modularized, high-speed data handling systems with interactive capability. Non-real-time bulk data processing will be provided by a separate suitable facility.

Plasma control is defined as feedback control on the plasma parameters such as plasma density, temperature, position, shape, fuel mixture, and volume. The plasma position control has been addressed on present machines. However, it will be different on TFCX because the structures will have nonuniform induced currents. Thus, the control field penetration will be very much nonlinear from a spatial consideration, and the induced fields or currents will be much larger in magnitude. The development of controls for regulating such items as plasma shape and plasma temperature will require a development program. The methods of measuring the shape and current densities will require future component development. The method of controlling the plasma temperature and power output for a 300-s burn has not been selected.

The stabilization of the elongated plasma before the stage of ignition is reached may require development. The elongated plasma is much more conditionally unstable in the vertical direction than for the case of the nonelongated plasma. Several options are available.

12. FACILITIES

The plant facilities support the operation and maintenance of TFCX. These facilities house and provide a controlled environment for the TFCX equipment. A suggested site arrangement is given in Fig. 12.1.

The emphasis of the facilities effort for TFCX has focused on devising the necessary facilities to support a TFCX design at a new site. In addition, most of the effort has focused on the reactor building and hot cell complex. The remaining facilities have been adapted from the earlier FED studies with appropriate modifications to reflect the change in mission.

A primary feature of the TFCX facilities layout is that the reactor hall (RH), the decontamination and hot cell facility (HCF), the tritium processing area (T), and the radioactive waste processing area (RW) all have been included on a common foundation mat. Figure 12.2 is a plan view of the TFCX nuclear island at two elevations (ground level and at 11 m). This feature provides a number of advantages, including shared common walls, simplified and improved seismic response, enhanced radiological safety by providing a single point of access to all potential radiation hazards, and concentration of the air-detritiation system into a single structure. The result is a much improved concept that is simpler and less costly.

TFCX FUSION ISLAND A

1. CO-2 STORAGE TANKS	11. CO-2 STORAGE TANKS	21. CO-2 STORAGE TANKS	31. CO-2 STORAGE TANKS
2. CO-2 STORAGE TANKS	12. CO-2 STORAGE TANKS	22. CO-2 STORAGE TANKS	32. CO-2 STORAGE TANKS
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8. CO-2 STORAGE TANKS	18. CO-2 STORAGE TANKS	28. CO-2 STORAGE TANKS	38. CO-2 STORAGE TANKS
9. CO-2 STORAGE TANKS	19. CO-2 STORAGE TANKS	29. CO-2 STORAGE TANKS	39. CO-2 STORAGE TANKS
10. CO-2 STORAGE TANKS	20. CO-2 STORAGE TANKS	30. CO-2 STORAGE TANKS	40. CO-2 STORAGE TANKS

Fig. 12.1. TFCX Fusion Island.

TFCX PLOT PLAN

Fig. 12-3. TFCX plot plan.

13. CONCLUDING REMARKS

The design and trade study information presented in this document represents a strong basis for understanding the superconducting and superconducting-copper hybrid TF coil options as candidates for TFCX. These designs are based on many years of development in reactor studies performed in the United States and around the world. They make use of the latest thinking regarding component design of the major systems and components, in particular, the design of the magnet systems, the torus systems, the impurity control systems, the plasma heating systems, and the facilities. The physics assumptions regarding confinement scaling are subject to improvement as new experimental information becomes available. However, the latest physics information has been incorporated into the TFCX design.

This interim document summarizes design issues, analysis, trade studies, and an ongoing effort to present a total, integrated, self-consistent design for the superconducting TF coil options for TFCX. Subsequent documentation will update the design evolution.

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