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Dr. 1858

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OAK
RIDGE
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LABORATORY

R & D Needs Assessment for the Engineering Test Facility

ETF Design Center Team

MASTER

Engineering Test Facility
Design Center

OPERATED BY
UNION CARBIDE CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

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R&D NEEDS ASSESSMENT FOR THE ENGINEERING TEST FACILITY

ETF Design Center Team

Date Published: October 1980

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ABSTRACT

The Engineering Test Facility (ETF), planned to be the next major U.S. magnetic fusion device, has its mission (1) to provide the capability for moving into the engineering phase of fusion development and (2) to provide a test-bed for reactor components in a fusion environment. The design, construction, and operation of the ETF requires an increasing emphasis on certain key research and development (R&D) programs in magnetic fusion in order to provide the necessary facility design base. This report identifies these needs and discusses the apparent inadequacies of the presently planned U.S. program to meet them, commensurate with the ETF schedule.

1. INTRODUCTION

In September 1978 the U.S. Department of Energy (DOE) issued a policy statement for fusion energy.¹ This policy statement encompassed the efforts in both the magnetic confinement and the inertial confinement approaches. In order to develop fusion energy as an economically attractive and environmentally acceptable energy option, the DOE policy statement on fusion energy outlined a three-phase strategy. Sequentially these phases focus on scientific feasibility, engineering testing, and reactor demonstration.

It is anticipated that scientific feasibility will be achieved within the 1980's for both the magnetic confinement approach and the inertial confinement approach. Following the achievement of scientific feasibility, each of the two approaches would move from the applied research phase into an engineering testing phase. The vehicle by which the fusion program would move into this phase of development is designated the Engineering Test Facility (ETF). The ETF would provide a test-bed for reactor components in a fusion environment that would be the essential building blocks of the facilities constructed and operated during the reactor demonstration phase. Commercialization would follow the demonstration phase.

In anticipation of the success of the near-term scientific and technological programs associated with the tokamak concept, an industry/laboratory design team has been assembled at Oak Ridge National Laboratory (ORNL). This team, designated the ETF Design Center Team, has as its objective to define a tokamak-based ETF and to prepare its design, schedule, and cost in appropriate detail to support the initiation of the ETF project at the earliest possible date. In order to guide the Design Center, an ETF Mission Statement² based on community-wide input was developed. As a point of departure for the design studies, a baseline design was adopted based on the TNS (The Next Step) studies³ of 1977 and 1978. The ETF Preconceptual Design Phase centers around two designs representing options relative to the Baseline Design. The Preconceptual Design Phase, initiated in FY 1979, will continue for approximately 18 months. The goal of this phase is to identify a single design that would be the basis for the Conceptual Design Phase.

As part of the Design Center activities, an ETF research and development (R&D) needs assessment was initiated at the same time as the Baseline Design specifications. The purpose of this needs assessment is to identify those technical areas that (1) are essential to the success of the ETF and (2) are not adequately covered in the current fusion programs. The purpose of this report is to present the results of the ETF R&D needs assessment.

Early in 1979 the International Tokamak Reactor (INTOR) study was established by the four major world fusion blocs: the U.S.A., the European Community, the U.S.S.R., and Japan. The initial task of the INTOR activity was to perform an R&D needs assessment. In the interest of avoiding a duplication of effort, the ETF and INTOR management agreed to share the first outputs from the teams of experts established for this purpose. The INTOR reports issued late in 1979 have proven very valuable in the Design Center needs assessment.

The R&D needs described here are paced to the ETF program schedule seen in Fig. 1.1 and are based on the ETF Baseline Design parameters listed in Table 1.1. An acceleration in the ETF program schedule would require an increased pace in the supporting developments while a delay in the schedule would permit a more moderate pace in the R&D. This assessment will be updated periodically to reflect accomplishments in the scientific and technological areas and to respond to any changes in the ETF schedule. A major goal of these assessments is to provide to DOE's Office of Fusion Energy (OFE) an identification of needs and recommendations for additional or accelerated R&D.

The body of this report (excluding the introduction, summary, and appendices) is divided into five main sections, one for each ETF Design Center branch area (Fig. 1.2). Each of these sections is further subdivided into subsections covering each of the topics listed in Table 1.2. All of the topics discussed in each branch area section are summarized at the beginning of that section. The report summary, Sect. 2 of this report, provides an overview. Relative priorities and need times have been identified throughout, and the major problem areas as now perceived are highlighted in Sect. 2.

Fig. 1.1. ETF FY 1984 line item project.

Table 1.1. Parameters for the ETF Baseline Design

Geometric

Plasma radius, a	1.2 m
Plasma elongation, σ	1.6
Aspect ratio, A	4.17
Major radius, R	5.0 m
Plasma edge-to-winding distance, Δ	1.28 m

Plasma

Beta, $\bar{\beta}$	7.0%
Plasma temperature, $\bar{T}$	12 keV
Ion density, n	$2 \times 10^{20} \text{ m}^{-3}$
Impurity indicator, Z_{eff}	<1.5
Safety factor, q	3.8
Plasma current, I_p	5.0 MA
Burn time (steady-state), τ_B	500 s
Duty cycle, dc	89%
Cycle duration	560 s
Energy confinement time, τ_E	1.2 s
Pellet fueling rate	$4.6 \times 10^{22} \text{ particles/s}$
Beam power into plasma, P_{inj}	50 MW
Beam energy, E_{inj}	150 keV
Microwave power, P_{rf}	1 MW
Microwave frequency, f	120 GHz
Plasma volume, V_p	227 m^3
Fusion power density, P_d	5.0 MW m^{-3}
Power (fusion), P_f	1140 MW
Total power (22.4 MeV/event^a), P_{tot}	1450 MW

Electromagnetic

Toroidal field on axis, B_T	5.3 T
Maximum toroidal field at coil, B_m	10.9 T
Number of TF coils	12
Magnetic field ripple at plasma edge, δ_a	1.5% peak to average

Table 1.1 (continued)

Electromagnetic	
TF coil type	Superconducting (Nb ₃ Sn)
OH solenoid type	Superconducting (NbTi)
PF coil type	
Inside TF bore	Copper
Outside TF bore	Superconducting
Total volt-seconds, ϕ_{vs}	83
Magnetic field in OH solenoid, B_{OH}	8.0 T
Thermal	
First-wall coolant	Water
Shield coolant	Borated water
Refrigeration load (TF coils), P_{ref}	48 kW at 5K
Power to first wall, $P_{\alpha/2}$	113 MW
Power to divertor, $P_{\alpha/2}$	113 MW
Power to shield, P_s	1224 MW
Plasma edge particle loss rate	3×10^{23} particles/s
Nuclear	
Neutron wall loading, L_w	2.4 MW/m ²

^a22.4 MeV/event assumed.

Fig. 1.2. ETF Design Center branch areas.

Table 1.2. Areas assessed for R&D needs

Plasma system
Energy and particle confinement
Heating physics
Startup, burn, and shutdown
Stability control
Impurity control, ash removal, and fueling
Nuclear systems
First wall and first-wall attachments
Blanket
Shield and nuclear analysis
Plasma chamber vacuum pumping
Fuel-handling and tritium considerations
Remote maintenance
Divertor collector
Electrical systems
AC power and energy storage
Neutral beam systems
RF heating
Diagnostics/instrumentation
Process controls
Fueling
Data acquisition
Magnetic systems
Toroidal field
Induction systems
Control and shaping coils
Divertor system
Cryogenic systems
Systems engineering
Reliability/availability
Maintenance philosophy
Materials

As a final point we note that many of the needs first determined appeared, after closer consideration, to be ETF design specific. Such needs were subsequently deemed to be research and development activities (RDAC's) candidates to be done as part of the ETF project rather than as part of the more generic R&D program. The development of tools to provide remote support to a specific ETF assembly procedure would represent a RDAC need. Generic R&D, on the other hand, would be represented by the feasibility demonstration of a plasma startup assist technique such as microwave heating. RDAC's will be specified and discussed in later documents and will be included in the plans and schedules for the design and fabrication of the ETF.

REFERENCES

1. John M. Deutch, *The Department of Energy Policy for Fusion Energy*, DOE/ER-0018, Directorate of Energy Research, Department of Energy, Washington, D.C. (1978).
2. ETF Design Center Team, *Mission Statement for the Engineering Test Facility*, ORNL/TM-6732, Oak Ridge, Tennessee (1979).
3. The results of the ORNL TNS Program studies are contained in ORNL/TM-6720-6731, Oak Ridge, Tennessee (1979).

2. SUMMARY

Both the scientific and technological data bases for tokamak development are expected to continue to expand rapidly as results from the current and planned programs accumulate. However, lack of knowledge and inadequate depth of understanding in some areas lead to the specification of the particular R&D needs identified in this document. In some cases the present information shortfall is expected to be eliminated by current programs at a pace commensurate with the ETF schedule. However, many R&D areas are not receiving the attention necessary to provide properly time phased answers for the ETF design evolution.

The ETF R&D needs are defined in the context of a design that responds to the two major elements of the ETF mission¹ (see Fig. 2.1): (1) physics issues, namely a burning, long-pulse, controlled plasma, and (2) facility issues, namely flexibility for testing, availability, and operator and public safety. The major physics issues tend to be feasibility concerns while the facility issues are those of operations and engineering design. In Tables 2.1-2.5 the essential R&D needs are arranged according to their priority within each branch area. All of the specific elements of ETF design and operation covered here are high-priority issues that (1) now present significant uncertainties, (2) must be resolved before the beginning of the Title I Design Phase, and (3) are inadequately covered in the present and planned programs.

It is apparent that many of these issues have some commonality and can be addressed by a smaller number of well-integrated, multi-issue R&D efforts. There are five such candidate R&D programs that can provide results for a large part of the needs. Thus, it is suggested that consideration be given to establishing integrated investigations as outlined in Table 2.6. Such dedicated R&D efforts in tightly coordinated program structures should ensure improved efficiencies and a better focus of the work to provide the required design data base for the ETF.

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THE R&D NEEDS ARE BEING DEFINED IN THE EVOLUTION
OF THE ETF DESIGN FROM THE MISSION STATEMENT

Fig. 2.1. ETF R&D needs defined in the context of a design that responds to the two major elements of the ETF Mission Statement.

Table 2.1. Major R&D needs of plasma system

Specific element	Degree of uncertainty	Potential impacts on ETF
Impurity control and ash removal	Large	Burn characteristics, availability, tritium and particle collection and pumping
Plasma disruptions control	Large	Designs and controls to limit disruption damage to walls and coils
Heating scenarios	Significant	Plasma control, energy of beams, reliability/availability
Preheating	Large	Cost and size of energy storage, first-wall designs, pulsed coils
Plasma reproducibility	Large	Availability of desired plasma conditions for testing

Table 2.2. Major R&D needs of nuclear systems

Specific element	Degree of uncertainty	Potential impacts on ETF
Divertor trapping of particles	Large	Long-pulse feasibility, particle processing
Tritium processing and control	Significant	Safety, cost
First-wall armor	Significant	Availability, designs for disruptions, operating life
Low-Z chamber coating	Significant	Impurity control methods, operating life
Torus sector seals	Significant	Maintenance concepts, availability

Table 2.3. Major R&D needs of electrical systems

Specific element	Degree of uncertainty	Potential impacts on ETF
Neutral beam systems	Moderate	Reliability/availability, cost
Pellet fueling and tritium handling	Moderate	Safety, cost
RF systems	Significant	Reliability/availability of power tubes and transmission

Table 2.4. Major R&D needs of magnetic systems

Specific element	Degree of uncertainty	Potential impacts on ETF
Divertor magnetics	Large	Remote handling, availability, ripple
Nb ₃ Sn at 12 T	Large	Physics margin, configuration for protection, ripple
OH coils (>100 MJ, ±7 T)	Significant	Reactor relevance, startup scenario, protection
Internal coils (insulators, remote handling, segmentation, disrupters)	Large	Availability, remote maintenance

Table 2.5. Major R&D needs of systems engineering

Specific element	Degree of uncertainty	Potential impacts on ETF
Reliability/availability	Large	Operating costs, test-time availability
Remote maintenance	Significant	Design complexity and costs, machine downtime
Materials	Significant	Cost and life of hard-to-maintain elements

Table 2.6. Integrated programs to meet R&D needs

Suggested programs	Specific ETF elements impacted
Impurity control and ash removal	Torus configuration TF coil configuration Ash removal/burn time Divertor magnetics Particle collection Tritium handling Materials
Disruptions (severity and frequency)	First-wall designs TF and PF coil protection Instrumentation and controls Reliability/availability Materials Controlled shutdown
Plasma control	Divertor system Disruption control/avoidance Controlled shutdown Pellet/gas fueling Pulsed superconducting coils and PF system
Plasma startup	Energy storage and power supplies Blanket/shield resistance Pulsed superconducting coils and PF system Startup assistance RF generation and launching
Plasma heating	Reliability/availability Remote maintenance Penetrations/shielding Beam energy

REFERENCE

1. ETF Design Center Team, *Mission Statement for the Engineering Test Facility*, ORNL/TM-6732, Oak Ridge, Tennessee (1979).

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3. PLASMA SYSTEM R&D NEEDS ASSESSMENT

3.1 PLASMA SYSTEM SUMMARY

The primary processes of interest discussed in Sects. 3.3-3.7 are listed for each of the physics areas in Tables 3.1-3.5. Each process has been rated on the criteria of degree of uncertainty, potential impact on the ETF, data needed before (date), anticipated data base, and priority in the plasma system considerations. Given the wide range and the difficulty of the physics problems confronting the ETF, the present tokamak program is doing a good job of addressing the majority of these problems.

Table 3.1 indicates the wide range of important problems associated with impurity control, ash removal, and fueling. Although many of these can have a large impact on the ETF, program coverage is in many cases marginal or inadequate. This area includes some of the most difficult physics problems for the ETF design and contains uncertainties in both engineering and physics requirements with currently prevalent impurity control methods. Their solution would provide a major contribution to the successful physics operation of the ETF.

The most important issue in the area of plasma stability (Table 3.2) is major plasma disruption. There are many important questions concerning disruption (time scale, energy deposition, plasma regimes free from major disruptions, etc.) that will greatly impact ETF designs. The present experimental program promoted the careful avoidance of plasma regimes prone to major disruptions. None of the major tokamaks in the U.S. is presently designed to handle the repeated major disruptions needed to carry out systematic investigations. Dedicated experiments on disruptions in a major tokamak that has been hardened to survive the repeated punishment of major disruptions are needed to provide necessary information on this process.

Heating scenarios using 150-keV beams and increasing densities should be tested in a large tokamak. Reliable, long-pulse (~ 5 -s) beam lines technology seems to be a pacing item. The reliance on complementary radio-frequency (rf) heating in the ETF depends on the demonstration of its physics feasibility (Table 3.3).

A number of issues involving startup, burn, and shutdown require emphasis. Fueling scenarios and burn control can have a large impact on the ETF, but the significant experimental tests will come during ETF operation. Theoretical studies and relevant experiments should provide marginal program coverage. Although preheating the plasma can impact ohmic heating (OH) coil and first-wall design by reducing the maximum voltage and the volt-seconds expended during breakdown, it has not yet received adequate experimental testing (Table 3.4).

In the area of energy and particle confinement (Table 3.5), the issue of plasma reliability and reproducibility is considered inadequately covered by the present tokamak program because the current emphasis is on scientific feasibility rather than on the statistical reproducibility of the achieved physics parameters. Plasma reproducibility is expected to have a large impact on overall ETF availability. The study of plasma reproducibility should also provide data in assessing the ETF design margin necessary to achieve reliable plasma operation. In a more general sense similar questions of reproducibility also apply to major R&D needs in other physics areas.

3.2 INTRODUCTION

The ultimate physics requirement of the ETF is the achievement of sustained controlled burning of fuel. Such an achievement will represent a large physics advance over the tokamak experiments of today. A number of component physics requirements must be met to achieve ignition and burn. These include

- (1) solving problems that could significantly limit the burn time such as control of impurities, fueling, and helium exhaust;
- (2) maintaining a stable plasma configuration that allows high beta and avoids major disruptions;
- (3) heating the plasma to ignition;
- (4) obtaining a successful operating scenario for startup, burn, and shutdown; and
- (5) adequately confining thermal and suprathermal particles and heat.

Although this division does an adequate job of encompassing the major physics problem areas, it should be emphasized that the areas are highly interrelated (e.g., particle confinement and impurity control or heating to ignition and ignition scenarios). The division of the physics area into these five groups was adopted by the INTOR participants, and R&D assessment was carried out in each area by an INTOR/ETF team of experts.

In Sects. 3.3-3.7 the ETF R&D assessment of each of these physics areas is presented. These assessments include descriptions of the area, the status of present programs, ETF time-phased needs, gaps between needs and projected available information, and recommended programs to fill these gaps. Some redundancy occurs because the areas overlap.

3.3 IMPURITY CONTROL, FUELING, AND EXHAUST

3.3.1 Introduction

The issues of impurity control, fueling, and exhaust (see Table 3.1) provide some of the most difficult physics problems for the ETF design. They are considered together because they could all have a major impact on achieving a long burn. Increased radiation in the core of the plasma because of an influx of impurities may quench the burn and make ignition difficult or even impossible. It will be necessary to fuel the burning plasma, or fuel depletion will limit the burn to 10-30 s. Similarly, fuel depletion may result from the accumulation of helium exhaust during the burn. Although the requirements for plasma purity, ash removal, etc., can be calculated reasonably well, achieving them may prove quite difficult.

An extensive analysis of the current status of the R&D needs in this area was recently completed by an INTOR/ETF team of experts chaired by R. Conn.

3.3.2 Impurity Generation

Impurities may be categorized by whether they are adsorbed on the wall (e.g., carbon, oxygen, nitrogen) or are the bulk material of the

Table 3.1. Impurity control, ash removal, and fueling

Processes of interest	Degree of uncertainty	Potential impact on ETF	Data needed before	Data base ^a	Priority in plasma system consideration
ETF tolerance to impurity and ash	Significant	Large	Title I	Adequate	Medium
Plasma and impurity transport in plasma edge	Significant	Significant	Title I	Adequate	Medium
Plasma and impurity transport in scrape-off region	Significant	Large	Conceptual/Title I	Marginal	High
Impurity generation	Significant	Large	Title I	Adequate	Medium
Nondivertor methods					
Removal and burial	Significant	Significant	Title I	Marginal	Medium
Plasma-edge modification	Significant	Significant	Title I	Adequate	Medium
Wall/limiter configuration	Significant	Significant	Title I	Marginal	Medium
Poloidal divertors (segmented)					
Magnetics	Small	Large	Conceptual/Title I	Adequate	High
Unloading and shielding physics	Large	Large	Conceptual/Title I	Marginal	High
Perturbations to plasmas	Significant	Significant	Title I	Marginal	Medium
Bundle divertors (or partial divertors)					
Magnetics	Significant	Large	Conceptual/Title I	Marginal	High
Unloading and shielding physics	Large	Large	Conceptual/Title I	Marginal	High
Perturbations to plasmas	Significant	Large	Conceptual/Title I	Marginal	High
Ash removal					
Helium, tritium, and deuterium separation	Large	Large	Conceptual/Title I	Not adequate	High
Fueling					
Pellet penetration and ablation	Large	Significant	Title I/Operation	Marginal	High
Gas blanket	Small	Significant	Title I/Operation	Adequate	Medium

^aSee Sect. 3.8.

wall or limiter. Adsorbed impurities, which can be removed by baking or discharge cleaning, are probably a much less serious consideration for the ETF than the wall-material impurities (usually metals).

Wall materials become impurities primarily through sputtering (by charge exchange neutrals and heavy ions, etc.), arcing, and evaporation. Many sputtering coefficients are known over a range of incident-particle energies and surface temperatures. However, incident plasma fluxes are not known, as discussed in the confinement section. Arcing is not understood quantitatively because of its dependence on details of surface and plasma structure. If the plasma is maintained more than a few centimeters from the wall, it will effectively prevent arcing. These processes have a strong dependence on the plasma-edge temperature and density, as indicated by several recent experiments. Evaporation is a difficult process to analyze, again because it depends on the distribution of the heat flux at the boundary.

Program plans for present tokamak experiments such as the Impurity Study Experiment (ISX), Doublet III (D III), Macrotor, the Princeton Large Torus (PLT), and the Poloidal Divertor Experiment (PDX) include extensive impurity-generation studies. The ETF needs to be able to estimate reliably the number and types of impurities that will be generated for a variety of plasma configurations (poloidal divertor, bundle divertor, no divertor, cold gas blanket, etc.) and wall materials. The major gap encountered again and again is the lack of experimental information about long-pulse, deuterium-tritium (D-T) experiments. Recommendations are to (1) measure the plasma fluxes on present experiments at wall, limiter, and divertor targets; (2) understand the models for metallic impurity generation; and (3) accumulate data on arcing, chemical erosion of coatings, and sputtering yields with incident particles >10 keV.

3.3.3 Nondivertor Methods for Impurity Control

Discharge cleaning and gettering

Discharge cleaning and gettering have proven quite successful in reducing adsorbed impurities in present-day plasmas. To evaluate

gettering for the ETF it would be necessary to know (1) the effects of high-temperature surfaces and long pulses and (2) the hydrogen and helium release rates for various wall coatings.

Impurity flow reversal, gas blankets, and expanded boundary

Impurity flow reversal is a neoclassical effect resulting from poloidally asymmetric hydrogen gas injection. Preliminary demonstrations in ISX-A have been successful, and additional experiments are planned in FY 1980 on ISX-B and D III. The gas input required in the ETF to achieve impurity flow reversal is quite large (comparable to the hydrogen particle loss rate). Thus, fueling in the form of gas injection might be done in conjunction with impurity flow reversal. In assessing this technique for reactor environments, experiments in which flow reversal is driven by local recycling would be useful.

Gas blankets may cool the plasma edge and spread the plasma energy over the entire wall such that heavy impurities are not generated. In the Oak Ridge Tokamak (ORMAK) and in PLT blankets created by hydrogen injection have reduced high-Z impurities. Recently Ohyaabu has proposed expanding the boundary region of the plasma by magnetically extending the outermost flux surfaces to result in more effective edge cooling by low-Z impurities. The ETF may use either gas injection or a boundary layer dominated by low-Z impurity radiation to achieve a cold gas blanket. An investigation of the possibility of maintaining a cold, stable layer of impurities at the plasma edge could fill the gap in the knowledge required before committing the ETF to a gas blanket scheme.

Modification of transport at the edge

Another means of lowering temperatures at the plasma edge is to increase diffusion there and hence increase the recycling of low-Z impurities at the edge. Suckewer and Hawryluk have proposed using rf waves to do this and have made calculations for application to the Tokamak Fusion Test Reactor (TFTR). Estimates for INTOR/ETF indicate that with this technique, as much as 40 MW of power may be radiated away

at the edge. To impact the ETF design, experiments on high-temperature tokamaks that determine rf power requirements are necessary.

Vacuum pumping

A major problem with divertorless operation of the ETF is vacuum pumping that requires the neutralization of the plasma near the ports and pumping speeds perhaps as high as 10^7 liters/s. This pumping speed may be reduced somewhat if alphas accumulate at the edge. Another possibility is to pump only between shots and attempt helium trapping by the wall to increase the burn time. At the present, large gaps in knowledge about the effectiveness of helium trapping exist. R&D needs include (1) the investigation of helium trapping and release in surfaces as a function of fluence, energy, and surface temperature using ion beams with energies down to 20 eV and (2) the investigation of simultaneous hydrogen bombardment and helium trapping.

Limiter modeling

The heat flux to the limiter in divertorless operation will be very high (~ 100 MW), a situation which introduces problems in heat transfer, thermal stress, thermal fatigue, and sputtering. Because of the long pulse length, the power level should be no greater than $0.1\text{--}0.2$ kW/cm², making necessary a limiter surface that is $\sim 10\text{--}30\%$ of the plasma surface. This would require extremely accurate plasma shape and position control.

3.3.4 Divertor Methods for Impurity Control

Introduction

Successful divertor operation could alleviate several problems discussed in the last section. A divertor will reduce particle and heat flux at the plasma edge and shield the plasma from incoming impurities. The impact on the overall design, however, is quite large.

Poloidal divertors

A poloidal divertor has been employed successfully on JFT-2A (DIVA), and plans for PDX include divertor operation with a 1-keV plasma in 1979 and a 5-10-keV plasma in 1980. Primary design considerations for a poloidal divertor are (1) the effect of fields upon equilibrium and stability, (2) maintaining divertor configuration during startup, (3) scrape-off channel geometry, and (4) remote maintenance. The outstanding problems are (1) neutron shielding of the divertor coils, (2) the possibility that the divertor burial chamber may be in a neutron environment, and (3) the remote maintenance requirement that the divertor coils be demountable. Credible coil designs require a normal coil with joints and modest shield requirements or a modular superconducting coil that is remotely maintained. Major gaps in knowledge again pertain to the lack of long-pulse, high-temperature experience. ETF R&D needs are to (1) demonstrate impurity control for reactor-like parameters over many energy confinement times, (2) determine particle pumping and energy removal requirements for the burial chamber, and (3) determine the effects on equilibrium and stability.

Bundle divertors

Bundle divertors have several potential advantages over poloidal divertors. With a bundle divertor the divertor coils may be removed without a major disassembly. The burial chamber is external to the toroidal field (TF) coils, and multiple units may be used, allowing maintenance and regeneration during operation. Major disadvantages may prove to be stresses on the coils and induced ripple. However, creative designers at Princeton Plasma Physics Laboratory (PPPL) and ORNL are already addressing these problems, working to develop improved local divertors. Limited experimental information is available from the Divertor and Injection Tokamak Experiment (DITE), and ISX-B will have an operating bundle divertor in late CY 1980. Confidence in bundle (local) divertor operation for the ETF can come only from a long-pulse experiment with reactor-relevant parameters.

Heat flux at divertor plates

Heat flux problems for the divertor plates are similar to those for the limiter discussed in the last section. In this case, however, it may be possible to spread the heat over a larger area by tailoring the magnetic field or relying on some intrinsic plasma instabilities in the divertor chamber.

Pumping in the divertor chamber

Pumping here differs in two important ways from that in divertor-less operation. The plasma is directed toward the pump by the divertor, and a neutral pressure differential may be maintained between the chamber and the torus. However, helium pumping remains a key requirement in the divertor chamber. ETF needs include an investigation of helium trapping in solid surfaces and liquids and a determination of what pressure differential can be maintained between divertor chamber and torus.

3.3.5 Fueling

Pellets

Present machines are fueled by gas puffing, but there are potential advantages to using pellets, including (1) lower net fueling rate and particle losses, (2) the possibility of plasma profile control, and (3) the ability to penetrate the divertor scrape-off layer. Pellet fueling has been demonstrated on ISX-A, and further experiments are planned for ISX-B and PDX. The prospect for achieving or exceeding the parameters required for shallow fueling scenarios by 1983 is good. If deep penetration is required (implying injection velocities approaching 10,000 m/s), an expansion of the existing pellet injector program would be required.

Gas puffing

Although gas puffing is not well understood theoretically, it is expected to work on the ETF. Divertor operation, however, could prevent gas penetration through the scrape-off region. R&D needs issues here are those discussed in the particle confinement section (Sect. 3.7).

3.4 EQUILIBRIUM AND STABILITY CONTROL

Achieving a controlled and sustained burn in the ETF will require either avoiding or controlling macroscopic plasma instabilities. The problems associated with plasma instabilities and their control may be divided into three broad areas: (1) stability limits on beta, (2) plasma disruptions, and (3) equilibrium position and shape control (see Table 3.2).

3.4.1 Stability Limits on Beta

It is important to determine the stability limit on beta because a minimum beta value (which depends upon the ETF geometry) is required to reach ignition. Hence, the aspect ratio, minor radius, and plasma cross-section shape must be chosen such that beta at ignition is less than the stable beta limit. The estimated beta limit for the ETF Reference Design is ~6%.

At present the experimental beta limit for a D-shaped plasma with a height-to-width ratio of approximately 1.5 has not been established. Experiments on ISX-B, PDX, and D III will address this problem; ISX-B is already probing beta limits in nearly circular plasmas. Of the three theoretical groups studying the ideal MHD limit for designs similar to that of the ETF, only the ORNL group obtained the beta limit on or above 6% for D-shaped plasmas with $A \cong 4$. This suggests that shape and current profile requirements to achieve high beta may be severe.

There are two major gaps in information at this time: (1) virtually no contact has been made between theory and experiment in establishing stable beta limits for circular or noncircular cross sections, and (2) assuming that precise current and pressure profiles are required to

Table 3.2. Stability and equilibrium control

Processes of interest	Degree of uncertainty	Potential impact on ETF	Data needed before	Data base ^a	Priority in plasma system consideration
Stability limits on beta	Significant	Large	Conceptual/Title I	Adequate	Medium
Plasma disruption control	Large	Large	Conceptual/Title I	Not adequate	High
Equilibrium position and shape control	Small	Large	Title I	Adequate	Medium
Vertical stability control	Significant	Large	Conceptual/Title I	Marginal	High
Impact of pressure-driven instabilities	Large	Large	Title I	Adequate	Medium

^aSee Sect. 3.8.

achieve high beta, it is not known if the plasma will achieve these automatically (for example, because of a fortuitous effect of pressure-driven instabilities) so that external profile control mechanisms will not be required. The noncircular tokamak experiments mentioned previously are expected to fill these gaps adequately, at least for non-burning plasmas.

3.4.2 Plasma Disruptions

Major disruptions play an important role in the first-wall design, wall-cleaning requirements, plasma control requirements, vertical field coil placement, and TF coil design constraints. During a major disruption of a burning plasma, 100-200 MJ of plasma energy may be deposited on the first wall of an ETF device. If this occurs on a fast time scale, substantial wall material will be vaporized. Localization of this energy could be disastrous. In addition, net ampere-turns linking the TF coils must be maintained for at least several seconds following disruptions to avoid excessive $\dot{B}$ at the superconducting coils.

Although disruption-free operating regimes can be determined empirically, major plasma disruptions are seen in all of the present experiments. Disruptions occur primarily during the high-current and/or high-density operations that are expected to occur at the level where the ETF must operate to achieve ignition and burn. The possibility of using helical coils to feedback stabilize disruptions has met with preliminary success on TO-1. Also, there seem to be some preliminary experimental indications in PLT and ISX of the reduced probability of complete current disruption during intensive neutral beam heating. Although theoretical investigations suggest that resistive tearing modes play a major role in disruptions, numerical simulations have been done only for circular cross sections and fixed plasma current. Hence, the final and most important stage of a disruption has not been investigated.

There is nearly a one-to-one correspondence between ETF needs and gaps in information concerning major disruptions. These are (1) a reliable estimate of the number of major disruptions expected during the lifetime of the ETF as functions of the plasma parameters, (2) the

time scale on which the plasma current quenches (a question yet to be addressed by the elaborate theoretical models), and (3) the spatial distribution of the energy on the first wall during a major disruption.

All of these questions received attention at the June 1979 Tokamak Optimization Workshop in Frederick, Maryland. It has been pointed out that experimentalists are not eager to use their devices to systematically study (rather than avoid) disruptions. However, PDX may provide an exception to this observation. Because of the impact of disruptions upon ETF design, the possibility of an experiment dedicated to the study of disruptions should be explored. A number of further suggestions for future theoretical development include (1) the effects of high beta plasmas, (2) noncircular cross sections and realistic boundary conditions, and (3) the examination of the process of current quenching following tearing mode overlap. These problems are known to the theoretical community and, with the possible exception of the problem of current quenching, will probably be addressed as rapidly as is feasible.

3.4.3 Equilibrium Position and Shape Control

The final stability control issue to be discussed here is equilibrium position and control, an area that does not appear to present critical problems for the ETF design work. It has been demonstrated theoretically that the hybrid poloidal field (PF) coil system used in the ORNL TNS design can maintain equilibrium position and control over a wide range of plasma current profiles. Other designs are probably also acceptable. Experimental confirmation of the equilibrium control of D-shapes with a 1.5 height-to-width ratio with a variety of PF coil systems will be provided by ISX-B, PDX, and D III.

Another related question concerns the limiting distance between the plasma and the PF coils beyond which the plasma stability with respect to vertical displacement can no longer be controlled. Plasma elongations around 1.4 have been obtained in D IIA and ISX-B at low β values ($\sim 0.3\%$) with close-by PF coils. Experimental demonstrations of vertical position stability control at higher β and larger elongation values are needed.

3.5 HEATING PHYSICS

3.5.1 Neutral Beam Heating

The effectiveness of neutral beam heating is well established. However, further research is needed to determine the startup scenario and the neutral beam energy required for the ETF. In addition, there are technology requirements for long-pulse operation in a neutron environment that must be addressed (see Table 3.3).

Shallow (?) heating is involved in a number of proposed startup scenarios to reduce neutral beam energy requirements to the 150 keV range. However, the efficacy of this heating approach has not been established. With this approach the temperature profile at the plasma edge in the ETF will be much steeper than that studied heretofore in PLT. This steep temperature profile could lead to enhanced impurity generation and to significantly reduced heating efficiency. The effects of edge heating should be studied in major tokamaks.

The use of a variable-density startup scenario to enhance neutral beam penetration requires both edge heating during the end of the startup scenario and a rapid density buildup. Rapid density buildup with pellet fueling may not work because of enhanced pellet ablation during neutral beam heating. On the other hand, rapid density buildup from gas puffing may be hampered by the slow movement of particles to the center of the plasma and by the possible chokeoff of the neutral beam lines. Experiments that study rapid density buildup during neutral beam heating are under way in ISX. Similar experiments in a larger device would also be desirable.

Another means of reducing beam penetration requirements is to operate in the hot-ion ignited mode where $T_i > T_e$ and $T_i \geq 30$ keV on axis. Based on Alcator scaling for the electron energy confinement, this mode of operation can reduce to ~200 keV the neutral beam energy required for a centrally peaked deposition profile. A further reduction in the beam energy required for centrally peaked beam deposition can be obtained by assuming a favorable effect of elongation on transport. The combination of high-temperature hot-ion ignited mode operation and a

Table 3.3. Heating physics

Processes of interest	Degree of uncertainty	Potential impact on ETF	Data needed before	Data base ^a	Priority in plasma system consideration
Neutral beam heating					
150-200-keV D ⁺	Significant	Significant	Title I	Marginal	High
300-keV D ⁻	Small	Significant	Title I/Operation	Marginal	Medium
Plasma rotation	Significant	Significant	Title I	Adequate	Low
RF heating ^b					
LHRH	Large	Significant	Title I	Adequate	Low
ICRH	Significant	Significant	Title I	Adequate	Medium
ECRH	Small	Significant	Title I/Operation	Marginal	Medium
Fusion alpha particle heating					
	Large	Large	Title I	Marginal ^c	High

^aSee Sect. 3.8.

^bOnly one successful rf heating method is assumed to be needed to supplement neutral beam heating.

^cSignificant testing is expected in the ETF.

favorable effect of elongation can reduce the neutral beam energy requirement for centrally peaked deposition to ~ 150 keV. However, the favorable effect of elongation upon transport has not yet been established. Information on this effect will be forthcoming in a number of experiments. It should be further noted that the use of hot-ion ignited mode operation to achieve adequate neutral beam penetration increases the beta requirement for ignition (by $\sim 50\%$) and relies on good energy confinement in an ion temperature range that is far from that employed in present devices.

For adequate confinement of neutral beam ions in the presence of significant field ripples, the injection angle is expected to be more parallel. This will tend to induce plasma rotation in the ETF via momentum transfer. Present experimental observations on plasma rotation are by and large inconsistent with theoretical estimates. Further investigations are suggested, especially those incorporating field ripple effects.

If approaches such as edge heating, low-density startup, and high-temperature hot-ion ignited mode operation coupled with favorable effects of elongation on transport are not possible, it will be necessary to use other techniques to enhance the penetration of 120-175-keV positive ion beams. Suggested techniques are ripple injection and compression boosting. Although tests of ripple injection will be performed on ISX, these investigations will not provide a definitive test of ripple injection. Compression boosting, in which a low-density plasma is neutral beam heated and then compressed to a higher density, rests on relatively firm experimental ground because of the Adiabatic Toroidal Compressor (ATC) experiments; however, even here some additional experimental work would be useful.

3.5.2 Beam Technology

Because the effects of the neutron environment on the neutral beam systems could lead to considerable difficulty, they must be carefully considered. Radiation damage to the ion source or the accelerator structure may result in the need for the frequent replacement of components. If these problems are too severe, a possible solution is to use

a bending magnet to facilitate the placement of the source out of direct line of sight of the plasma; however, beam optics and control requirements will result in additional problems that need study. A further problem caused by the neutron environment may be the heating of the cryopanel; the use of other means of pumping should be examined. The possibility of increasing the amount of effort devoted to the study of neutron effects on neutral beam heating systems should be considered.

Another aspect of neutral beam technology development that should be considered is the possible need for steady-state beams. Steady-state operation would be needed for high-Q driven operation if ignition cannot be obtained or if ignition is terminated by impurity buildup or by minor disruptions. Steady-state high-Q driven operation may also be desirable in order to obtain thermal stability.

3.5.3 Lower Hybrid Heating (LHH)

Lower hybrid heating (LHH) experiments in Alcator indicate that the phase velocity in the parallel direction is slower than expected. This decrease in phase velocity will lead to strong Landau damping absorption at the plasma edge and prevent penetration of lower hybrid radiation to the center of a high-temperature plasma. Even in the absence of this effect, the absorption of lower hybrid radiation at the edge of the plasma due to Landau damping can be a problem during heating to ignition temperatures; therefore, special startup scenarios will be required. Major experiments in Alcator C and D III will play an important role in determining the applicability of lower hybrid heating to the ETF. Even if these experiments are successful, however, it will still be necessary to show that lower hybrid heating can be used to obtain high temperatures (≥ 10 keV).

3.5.4 Ion Cyclotron Range of Frequencies (ICRF)

The results of recent ICRF experiments on PLT are encouraging. In addition, there are a number of heating experiments planned for the next few years (on PLT, Alcator C, Macrotron, TFR, and T-10) that should be

able to provide the information needed to assess the applicability of this approach to the ETF. However, even if these experiments are successful, further experiments that study heating to higher temperatures may be necessary. Also, further effort is needed in the development of effective launching structures. Problems of neutron and plasma heat loading of the launching structure must be investigated.

3.5.5 Electron Cyclotron Resonance Heating (ECRH)

The absorption of radiation in ECRH should take place by well-understood classical processes. The efficacy of ECRH has been established in Soviet experiments. Future experiments are planned in the Soviet Union and the U.S. The major question in determining the appropriateness of ECRH for bulk heating in the ETF is whether the gyrotron sources of high-power radiation in the 150-200-GHz range can be developed. Furthermore, if the ETF operates at significantly different values of magnetic field during different phases of operation, it will be necessary for the gyrotron to have a two-frequency capability; if this capability does not exist, it will be necessary to have two sets of tubes. The development requirements for gyrotrons can, of course, be significantly reduced if ECRH is used for initial plasma startup and profile control but not for bulk heating.

3.5.6 Alpha Particle Heating

It is highly probable that experimental studies of ignition physics, thermal stability, and burn control will not be made before ETF operation unless a copper magnet ignition test reactor such as the Zephyr device being discussed at Garching is operated. TFTR and the Joint European Torus (JET) will provide some information on alpha particle heating but are unlikely to provide definitive information on profiles of self-heated plasmas, alpha-particle-induced instabilities, disruptions in ignited plasmas, tradeoffs between ignition requirements in various regimes of ignited operation (e.g., low temperature vs hot-ion mode), and the efficacy of various burn control techniques. Increased support

will be needed for theoretical studies of alpha particle physics and for plasma engineering related to ignition to bridge the gap between TFTR/JET and the ETF.

3.6 STARTUP, BURN, AND SHUTDOWN

Consideration of the complete plasma discharge (startup, burn, and shutdown) necessarily involves physics issues detailed in the other R&D assessments in this section. For example, an assessment of heating to ignition requires extensive knowledge of energy and particle confinement as well as auxiliary heating characteristics. Here we attempt to concentrate on issues not directly covered elsewhere (see Table 3.4).

3.6.1 Startup

The startup scenario, the preignition phase of the discharge, will have a strong impact on the ETF design because it determines PF energy storage and power supply requirements, heating and fueling requirements, and, to a lesser extent, plasma equilibrium control systems. It is useful to divide the startup phase into (1) current initiation of the discharge and (2) heating to ignition.

No large problems are anticipated in the first stage of the discharge but at least two issues deserve mention. The first concern is plasma preheating. It may be possible to reduce dramatically the number of volt-seconds as well as the peak voltage required in the induction coil circuit by adopting a preionization and preheating technique such as the upper hybrid resonance heating. However, this is a speculative procedure and should be explored experimentally as planned on ISX-B. If that experiment proves successful, experiments on larger devices would then be useful.

Another concern in the early history of the discharge is current penetration. Although experiments performed on PLT indicate that enhanced current penetration does occur, there is some evidence that impurity influx is associated with the MHD activity accompanying this anomalous current penetration. Therefore, an expanding radius scenario that could

Table 3.4. Startup, burn, and shutdown

Processes of interest	Degree of uncertainty	Potential impact on ETF	Data needed before	Data base ^a	Priority in plasma system consideration
Preheating	Large	Large	Conceptual/Title I	Not adequate	High
Current buildup	Significant	Significant	Title I	Adequate	Medium
Heating scenarios	Significant	Large	Title I	Not adequate	High
Fueling scenarios	Large	Large	Title I	Marginal ^b	High
Burn control	Significant	Large	Title I	Marginal ^b	High
Burn quench	Significant	Significant	Title I	Marginal ^b	Medium
Current quench	Significant	Significant	Title I	Adequate	Medium

^aSee Sect. 3.8.

^bSignificant testing is expected in the ETF.

alleviate current penetration problems and mitigate the generation of impurities may be desirable. Experimental tests on PDX and TFTR will be useful.

The second stage of startup is heating to ignition. We defer the relevant assessments of transport, heating mechanisms, and impurity control to other discussions in this section. Attractive neutral beam heating scenarios (employing beam energies of 150-200 keV) exist, provided one can start at low densities ($n = 5 \times 10^{13} \text{ cm}^{-3}$), thus achieving full beam penetration, and then fuel concurrently to raise the density. It must be stressed that this fueling entails more than just raising the density; the proper D-T fuel mixture must be spatially maintained. This area has received little theoretical or experimental attention. A theory for gas puffing and particle diffusion that has predictive power is required; none exists at the present time. Experiments that puff hydrogen into machines previously running with deuterium have shown that the resulting fuel mixture is overwhelmingly influenced by the wall gas inventory. Further experiments in diverted plasmas should be undertaken.

3.6.2 Burn

The ETF can meet its reactor-relevant objectives only if a sustained, reproducible plasma burn is achieved. Major considerations for this phase of the discharge are the pulse length and the control of the burn.

A number of factors determine the burn pulse length including volt-second limitations, decay of flux-conserved high-beta equilibria, impurity generation and accumulation, and alpha particle accumulation. Here we assume that the impurity problem has been solved (see Sect. 3.3 on impurity control). Of the remaining factors, the alpha accumulation problem is almost certainly the most restrictive. Experimentally this problem will not be fully assessed until D-T operation of TFTR and JET begins. Theoretically we have the common problem of the lack of a reliable model for particle diffusion. A zero-dimensional (0-D) model employing perfect helium recycling at the wall predicts a burn length of

23 s for typical ETF parameters. Assuming the energy and particle confinement times for D-T and alphas are comparable, a recycling coefficient of 96-98% may be small enough to reach a steady-state burn. These results are highly model dependent, and a meaningful 1-D simulation is difficult because of the large uncertainties associated with particle transport; much further work is needed.

A burn control mechanism will be required to sustain the plasma burn because the ignition point in an axisymmetric ETF plasma is expected to be thermally unstable and because a temperature excursion is expected to follow. If loss mechanisms with a temperature dependence stronger than $\langle \sigma v \rangle_{DT}$ do not appear, then eventually beta will exceed the MHD limit, with yet unknown consequences. If confinement turns out to be a more severe constraint than the beta limit, then the possibility of stable burn exists in operating at low densities and high ion temperatures ($T_i \sim 40$ keV), the "hot-ion mode." In this temperature regime, $\langle \sigma v \rangle_{DT}$ varies weakly with the temperature, and bremsstrahlung is sufficient to ensure a minimal temperature excursion to a stable burn point. However, very stringent ripple requirements will be necessary to achieve ignition in this regime. The effects of ripple on achieving high temperatures are discussed elsewhere in this report. In the following discussion we assume the standard $T_i \cong 10$ -15-keV approach, which may be consistent with practical levels of ripples.

An attractive candidate for a burn control mechanism is enhanced transport due to MHD instabilities when the critical beta value is exceeded. Current experimental programs (D III, ISX-B, and PDX) that are to explore beta limits can also examine the effects on confinement of exceeding the theoretical beta limits. This would provide valuable information in assessing the possibility of using MHD instabilities as a burn control mechanism.

Fueling is another possible burn control mechanism; for example, one may optimally vary the D-T fuel mixture. However, for a variety of reasons controlled fueling is not an attractive option. The particle transport process may not permit lowering the density on the thermal excursion time scale; staying near the ignition point requires low

tritium density and hence low thermonuclear power, etc. The feasibility of this option will be rather difficult to determine prior to D-T operation in TFTR and JET.

The final burn control mechanism considered here is enhancing the ion thermal conductivity with TF ripple. Because of the strong temperature dependence of thermal diffusivity on this loss mechanism ($\propto T_i^{7/2}$), thermal excursions may be prevented. Although the results of preliminary modeling studies are promising, potentially severe problems have been identified. For a significant ripple at the plasma center ($\sim 1\%$ peak to average), hot alphas are preferentially lost, thus reducing fusion plasma heating at the center and increasing the high-energy flux on the first wall. Also, the ripple operating window that allows both ignition and an acceptable burn point may become uncomfortably small, particularly considering the uncertainties in ripple transport. However, the use of compensating ripple coils may alleviate this problem. Theoretical and experimental understanding of ripple-induced transport is needed, as discussed in Sect. 3.7.

3.6.3 Shutdown

The shutdown (discharge termination) phase consists of quenching burn and then extinguishing the plasma current without inducing major disruptions. Of course, a controlled shutdown is preferable, perhaps obtainable with a burn control technique that is applicable over a wide enough parameter range.

Adequate specification of the ignition control and shutdown systems in the ETF will require experiments to expand our data base in several areas. The character of the systems will depend strongly on the nature and parameter dependence of the transport near the beta limit. Use of fuel control for burn termination will require data concerning particle transport coefficients, particularly near the plasma center. If ripple-enhanced alpha loss is employed, further experimentation will be required to determine the ripple/alpha interaction. A good experimental study of the effect of ripple on alpha confinement could be performed on TFTR and/or simulated in smaller machines with injected energetic particles.

If burn control is dominated by the MHD beta limit, burn quenching may be effected by deliberately reducing the beta limit in a controlled fashion. Whether this is done through changes in shape or other plasma parameters will become clear when experimental tests of beta limits in ISX-B, PDX, and D III are understood.

After ignition has been quenched, the plasma current must be reduced to zero without inducing current disruptions. In principle the induced skin currents during termination can be controlled by scraping the plasma surface on the limiter. However, this technique has not been adequately tested experimentally and may lead to disruption. In addition, there is a strong probability that the plasma density maintained during ignition will be too high for the OH input according to the Murakami scaling. In this case, current channel contraction and associated disruption may occur unless plasma density is also significantly reduced when the burn is quenched. If this is found to occur during injection-heated experiments planned for the near future, the implications for the ETF will have to be evaluated.

3.7 PLASMA ENERGY AND PARTICLE CONFINEMENT

3.7.1 Introduction/Description of Area

This area deals with the ability of the magnetic field configuration to contain particles and energy so that significant fusion reactions may occur. The condition under which a plasma with sufficient density n and temperature T is contained over a time period τ to permit self-sustained fusion burn can be represented by the product of $Tn\tau$. For D-T fusion $Tn\tau$ must be greater than a few 10^{15} s cm⁻³ keV.

Several processes are believed to degrade confinement from the fundamental level of the neoclassical transport. These processes may involve instabilities, impurities, imperfections in magnetic fields, neutral particles, etc. An understanding of these processes has been of major interest in the U.S. magnetic fusion program. In the case of tokamaks a significant amount of information and insight has already been obtained through the combined effort of experiments and applied

plasmas physics studies. However, a number of critically important questions still remain; these will be discussed below (see Table 3.5).

The following discussion concerns plasmas without significant impurity accumulation and macroscopic instabilities, but with possible field imperfections, neutral particles, and turbulence of microscopic instabilities. The R&D assessments in the areas of impurity and stability control are discussed in Sects. 3.3 and 3.4 and are concerned with methods to ensure that only insignificant levels of macroscopic instabilities and impurities remain. Questions concerning the physics of auxiliary plasma heating to ignition are examined under plasma heating (Sect. 3.5). The present ETF R&D assessment has incorporated the U.S. INTOR reports on various physics issues and the earlier TNS R&D needs assessment reports.¹

3.7.2 Present Status of Program

Energy confinement

Thermal conduction. Energy confinement studies in a large number of tokamaks have revealed that the simple rule (Alcator scaling) $\tau_E = 5 \times 10^{-19} \bar{n} a^2$, with units in s, cm^{-3} , and cm, applies to ohmically heated plasmas to within a factor of 2. Evidence from Alcator and ISX-A further indicates that at high density and over the central part of the plasma, τ_E becomes increasingly dominated by the ion neoclassical thermal conduction. It can be inferred from these and other evidences that the electron thermal diffusion coefficient varies as $\chi_e \sim (5 \times 10^{17}/n_e)$. The high ion temperature ($T_i \sim 6$ keV) achieved in PLT has indicated that the trapped-ion mode may not be deleterious to the ion energy confinement at the low collisionality expected of the ETF. There are reliable measurements in ohmic-heated and injection-heated tokamaks such as PLT, ISX-B, Alcator A, etc., which have indicated that χ_i is within a factor of 2-3 above what the neoclassical theory indicates.

The processes involved in the electron heat conduction are still quite uncertain. In the presence of dominating auxiliary heating such as in PLT and ISX-B, indications of a favorable T_e dependence in χ_e have

Table 3.5. Energy and particle confinement in plasma core

Processes of interest	Degree of uncertainty	Potential impact on ETF	Data needed before	Data base ^a	Priority in plasma system consideration
Ion thermal conduction	Significant	Modest	Title I	Adequate	Low
Electron thermal conduction	Large	Large	Conceptual/Title I	Adequate	High
Particle transport and flux	Large	Large	Title I	Adequate	High
Ripple effects	Large	Large	Conceptual/Title I	Marginal	High
Impurity transport in plasma	Significant	Significant	Title I	Adequate	Medium
Neutral effects	Significant	Significant	Title I	Adequate	Medium
Plasma reliability	Large	Large	Conceptual/Title I	Not adequate	High

^aSee Sect. 3.8.

been obtained in addition to the n_e dependence in the Alcator scaling. In an ETF-sized plasma, the presence or absence of the T_e dependence may result in an order of magnitude change in χ_e . The theoretic underpinning of the observed χ_e scaling is not yet available. Recent work by Hirschman and Molvig on drift-wave turbulence in tokamaks may be a significant advancement in this area.

Ripple. Imperfections in the magnetic field configuration are expected to result in enhanced ion energy losses through the induced perturbations in the ion drift orbits. Several distinct ripple processes have been identified to be potentially important. These include plasma diffusion due to ripple trapping, ripple-induced banana orbit drift (called the ripple-plateau drift), and enhanced losses of suprathermal particles with a similar set of processes. The resulting ion thermal diffusivity is expected to have a strong dependence in ion temperature (or energy) of the form $T_i^{7/2}$ (ripple trapping) or $T_i^{3/2}$ (banana drift) and thus may have a strong limiting effect on the achievable T_i in the ETF. Experimental studies of this topic have so far been rather limited. Some early indications in ISX-B point to significant losses of suprathermal ions in the presence of an asymmetric ripple ($\sim 1.5\%$) at the plasma center without significant increases in thermal conduction. At the present time the uncertainty in the permissible magnitudes of ripple is significant.

Neutrals. The neutral particles in ohmically heated tokamaks are understood to be primarily near the plasma edge and to be a channel of significant energy loss there via charge exchange and radiation. At the plasma center, however, the neutral concentration is dramatically increased by intense neutral injection. For a given T_e and beam energy, the beam-introduced neutrals n_{BN} , the neutral beam power density p_B , the direct power loss through these neutrals p_{DL} , and the secondary charge exchange loss of the streaming beam ions p_{SCX} roughly scale as

$$n_{BN} \propto n_B n_p ,$$

$$p_B \propto n_B n_p / \tau_s \propto n_B n_p^2 ,$$

$$p_{DL} \propto n_{BN}/n_p \propto n_B, \text{ and}$$

$$p_{SCX} \propto n_B n_{BN} \tau_s \propto n_B^2 n_p,$$

where τ_s is the slowing-down time and n_B and n_p are the neutral beam and plasma density, respectively. Intense injection (large n_B) into a plasma with sufficiently low n_p may eventually limit the net beam heating power ($p_B = p_{DL} - p_{SCX}$). This self-limiting effect on the effectiveness of neutral beam heating, not yet well understood in PLT and ISX-B, may have a strong effect on heating scenarios in the ETF.

Particle confinement

The understanding of particle transport processes in tokamaks is poor at the present time. A persistent puzzle yet to be understood is the apparently rapid inward transport of particles as they are introduced near the plasma periphery.

In ohmically heated discharges where the neutral density at the plasma boundary can be controlled, the maximum achieved density has been found to increase with B_T/R_0 (Murakami scaling) and decrease with the level of impurity. The value of B_T/R_0 can be linked to the net heating power input in ohmic discharges. Earlier ORMAK neutral injection experiments seem to have confirmed this scaling because the plasma density limit increased with the added beam power.

However, preliminary indications from ISX-B under intense neutral beam power have pointed to a possible variation to the density limit based on input power in that the density apparently tends to clamp at the preinjection values when large parallel injection power is introduced. The processes at work here may include the self-limiting effect of beam-introduced neutrals, MHD pressure-driven instabilities, beam/plasma instabilities, or an alteration of the rapid density-pinching process by intensive neutral injection, etc.

Given that the particle transport is a net result of inward and outward fluxes, the net particle confinement time can be defined as

$\tau_p(r) = N(r)/S(r)$, where S is the volumetric ionization source and N is the total number of particles within radius r . When $S(o)$ is significant in relation to $S(a)$, experiments indicate that $\tau_p(a) \sim 3\tau_E$ and, in some cases, $\tau_p \propto n$. When $S(o) \ll S(a)$, as in the case of a large tokamak with particle sources limited to the plasma periphery, $\tau_p(o) \gg \tau_p(a)$ is expected.

On the other hand, there is a clearer understanding of the confinement of suprathermal particles in axisymmetric tokamak configurations because of rather successful studies of the neutral injection physics in ORMAK, ATC, PLT, and ISX-B. The effects of field ripples on the suprathermal particles, however, have not yet been thoroughly studied. An understanding of this topic has straightforward application to the confinement of fusion alpha particles.

Impurity transport

Because the particles do not seem to conform to the neoclassical transport theory, it is not surprising to find similar discrepancies in impurity transport, although the impurity dynamics almost invariably resides in the collisional (Pfirsch-Schlüter) regime. Measurements of $Z_{eff}(r)$ in a large number of tokamaks have not resulted in a consistent picture of the Z_{eff} distribution. However, controlled experiments in Alcator and PLT with pulsed impurity injection and transport have revealed a rather consistent impurity diffusion speed of 1-3 cm/ms and impurity confinement times which appear to be close to that of the background plasma. That the impurity behavior nearly conforms to the neoclassical collisional transport is further supported by the preliminary indications of flow reversal experiments in ISX-A. Although significant questions on the transport processes of impurities still remain, these point to the strong likelihood that, if impurities are continuously screened and removed at the plasma boundary, the overall plasma purity can, in principle, be maintained.

Plasma reliability

An important measure of our knowledge of fusion plasmas is the reproducibility of a certain plasma state in an experimental device as a function of major plasma and device parameters (e.g., I_p , B_T^a , n , T , Z_{eff} , shape, etc.). The number of discharges achieving the experimental milestones in major tokamak devices has invariably been small compared to the total number of discharges attempted. Although successful operations of experimental tokamaks have been achieved, statistical studies on our ability to operate and control reliably the plasmas in major experimental devices have not yet been done.

This situation appears to be the result of several factors. The experimental program presently emphasizes proving the scientific feasibility of fusion-relevant plasma parameters. Component failures and the learning aspect of tokamak plasma operation have also discouraged the study of the reliability in producing desirable plasma parameters in major tokamak devices. The achieved level of reproducibility of plasma parameters in present-day devices is believed to be low compared to the desirable level in a device like TFTR and the ETF.

3.7.3 ETF Needs

If the ETF were to be designed and built using only our present knowledge of plasma confinement, significant conservatism would have to be incorporated into the design of the device to ensure its success. The resulting device would probably be large, costly, and unattractive. To make the ETF viable, therefore, we must reduce the uncertainties in our understanding of those confinement processes to a level that can be covered by relatively manageable margins in the design.

Ion thermal conduction

The magnitude of χ_i in the ETF based on neoclassical theory is estimated to be significantly smaller than χ_e based on Alcator scaling. If this trend continues, the uncertainty by a factor of 2-3 in χ_i is expected to have a modest impact on the ETF design. However, in order

to resolve the effects of impurities and neutral particles in analyzing experimental data, it is often necessary to make more accurate estimates of the ion thermal conduction process. Thus, as the experimental program progresses, a reduction in uncertainties of our understanding of χ_i can be expected. Confirmation of this nearly neoclassical behavior of χ_i in TFTR is necessary as input to the ETF Title I Design.

Electron thermal conduction

The ETF Baseline Design has plasma size and B_T estimates based largely on the assumption of $\chi_e \sim 5 \times 10^{17}/n_e$. An additional dependence of χ_e on $1/T_e$ is expected to result in a large increase in the plasma energy confinement time and to impact strongly the desirable plasma size and B_T . An increased certainty in this χ_e scaling is needed to allow for a credible conceptual design. Also, our understanding of the electron heat conduction processes needs to be increased (to within a factor of roughly 2) before the Title I Design can be carried out with sufficient confidence.

Particle transport and flux

The process of concern here involves how neutral beam injection and alpha particle heating can influence the net inward plasma flux needed to increase the plasma density during low-density heating to ignition. This uncertainty is expected to influence whether neutral beam energies ~ 150 keV and pellet speeds ~ 1 -2 km/s are indeed sufficient in achieving ignition. A resolution of this is needed before the Title I Design begins.

Ripple effects

Uncertainties about the limiting ripple magnitude are expected to restrict the size and number of TF coils and have a large impact on the engineering feasibility of the ETF. Needed are a preliminary (theoretical)

clarification of this issue before the Conceptual Design begins and an experimental resolution of the admissible limits of periodic and localized ripples before the Title I Design begins.

Impurity transport

Given that the impurities are successfully removed from the plasma boundary, the key question is whether or not they tend to accumulate at the plasma center in a fusion-grade plasma. A resolution of this uncertainty is also needed before Title I Design begins.

Neutral effects

As long as the neutral beam is the primary heating method to achieve ignition, the neutral particles it introduces will result in a significant perturbation in plasma confinement. This impact needs to be incorporated into the Title I Design.

Plasma reliability

The degree of reproducibility of the desirable plasma discharges will have a strong impact on ETF reliability and availability both during the hydrogen and D-T plasma operation testing phase and the engineering testing phase. There is presently no data base to support a credible estimate of the ETF plasma operation needs for the Conceptual and Title I Designs. It is necessary, therefore, to begin to establish statistics of plasma reliability in major tokamak devices.

3.7.4 Program Coverage and Gaps

Plasma confinement has enjoyed attention for many years in tokamak experiment and theory programs. Experimental and theoretical research activities related to PLT, PDX, D III, ISX-B, Alcator C, TFTR, and TEXT are considered adequate in reducing, in the next several years, the uncertainties in our understanding of the ion and electron thermal conduction, particle and impurity transport (in plasma), and neutral

effect. However, it is necessary to resolve the influence of T_e on χ_e relatively early in the Conceptual Design Phase (1981-1982) because of its potentially strong impact on the desirable ETF plasma size and B_T .

On the other hand, large gaps exist in the areas of ripple effects and plasma reliability. Presently only ISX-B has installed ripple injection coils and plans to add a bundle divertor, making it possible to examine the effects of localized ripples on plasma confinement. There is yet no experiment available to study the effect of significant periodic ripples in toroidal fields, although theoretical studies in this area have produced some useful results and estimates. However, significant controversy still remains. To obtain a credible data base on ripple effects in time to impact the ETF Conceptual Design, an emphasis in ripple experiments and theory is needed. Modifications of some existing tokamak devices to study the ripple effects seem necessary.

A new type of experimental activity is also needed to produce a data base on plasma reliability and reproducibility. Instead of stopping a small sequence of discharges that have achieved interesting parameters, it is necessary to carry out a larger number of discharges for sufficient data to define the uncertainties in the measurements and the reproducibility of plasma parameters. The reliability of the tokamak components involved in the experiments needs to be examined concurrently. When the plasma operating regime is changed, it is necessary to examine the causes of the unsuccessful discharge attempts before a successful sequence of discharges can be established. The resulting statistical data will provide a basis for assessing the implication of the large availability (25-50%) in the ETF.

3.8 1979 ESTIMATES OF U.S. TOKAMAK EXPERIMENTS: PHYSICS R&D DATA BASE COVERAGE FOR ETF

The following are input from knowledgeable tokamak experimenters of future U.S. tokamak experiments. The information is categorized with the physics processes that can strongly impact the ETF design and is used as the reference to assess the adequacy of the physics data base for the ETF. Special thanks are due to G. Bekefi, A. Gondhalekar,

D. Overskei, and R. Parker of the Massachusetts Institute of Technology (MIT); F. Marcus, T. Tamano, and J. Wesley of General Atomic (GA); J. Dunlap, R. Isler, T. Jernigan, J. Lyon, M. Murakami, J. Sheffield, and W. Wing of ORNL; D. Meade, J. Schmidt, and W. Stodiek of PPPL; R. Gross of New York University; R. Taylor of UCLA; G. Goldenbaum of the University of Maryland; R. Bengston of the University of Texas; and C. Sprott of the University of Wisconsin.

Useful suggestions from P. Stone and R. Blanken of DOE are also acknowledged and included.

3.8.1 Electron and Ion Thermal Conduction

Alcator C	December 1979	High-density, high- $n\tau_e$ plasma at low plasma temperature
D III	1979	Ohmic-heated D-shaped and doublet plasmas
	1981	7-MW injection-heated D-shaped and doublet plasmas
ISX-B	December 1979	High-beta circular plasma
	December 1980	High-beta noncircular plasma
	December 1981	High-beta noncircular diverted plasma
PDX	December 1979	D-shaped OH plasma
	September 1980	High-beta collisionless diverted plasma
PLT	1979-82	Circular tokamak plasma with neutral beam heating, ICRH heating, LH heating, ohmic heating
TFTR	June 1982	Circular TFTR plasma with ohmic heating
	December 1982	Circular TFTR plasma with beam heating
TERP		Confinement physics
Tokapole		Transport studies

3.8.2 Particle Transport and Flux

Alcator C	July 1980	Neutral buildup in low-temperature, large- n_e plasma
D III	1979	In situ pellets and gas puffing
ISX-B	June 1980	Gas puffing
	June 1980	Single pellets
	December 1980	Repetitive pellets
PDX	October 1979	Low-temperature diverted plasma with pellet injection
	September 1980	Beam fueling in high-beta collisionless plasma
PLT	1979-82	Parameter variation with heavy gas injection, single pellets and various auxiliary heating, measuring low-energy neutral spectrum, high-energy neutral spectrum, spectrum of heavy and light impurities and H_2 , D_2 , He
TFTR	June 1982	Ohmic-heated TFTR plasma
	December 1982	Beam-heated TFTR plasma

3.8.3 Ripple Effects on Particle and Energy Confinement

Alcator A	December 1979	Symmetric fixed TF ripple ($\sim 4\%$ near plasma edge)
ISX-B	December 1979	Asymmetric localized ripple
	December 1980	Periodic ripples (9 TF coils)
	December 1981	Bundle divertor ripple
TFTR	February 1983	Precompression hot-ion ripple interaction
	March 1984	Ripple/alpha interaction
PLT	January-June 1980	Single TF coil ripple (up to 2% on axis), TF ripple with coils in 6 groups of 3

3.8.4 Impurity Transport in Plasma

Alcator A	December 1979	Transport of iron, aluminum, molybdenum by pulsed techniques
Alcator C	1980-81	Effect of size on impurities (from Alcator A to Alcator C), impurities transport at large $n\tau$'s
D III	1979-80	Impurity flow reversal in D-shaped and doublet plasmas
ISX-B	1979-81	Transport of naturally occurring impurities (carbon, oxygen, iron) and externally injected impurities (neon, tungsten, etc.), with and without beam-driven momentum and with and without poloidally asymmetric particle sources
PDX	October 1979 September 1980	Transport of injected impurities Effect of boundary modification on impurity transport
PLT	1979-82	See "particle transport and flux"
TFTR	December 1983	Pellet-impurity injection results

3.8.5 Neutral Effects

Alcator C	1979-80	Gas puffing in high-density, large $n\tau$'s plasma
D III	1979-81	Intense neutral injection, gas puffing, in situ pellets
ISX-B	1979-81	Intense neutral injection, gas puffing, pellet injection
PDX	November 1979 June 1980	Effect of extensive gettering Studies of internal regenerable getters
PLT	1979-82	High-power neutral injection, gas puffing, pellet injection
TFTR	1983	Intense neutral injection, gas puffing, pellet injection

3.8.6 Plasma Reliability and Reproducibility

Alcator A	June 1980	Origin and characteristics of minidisruptions
ISX-B	1979-81	Statistical studies of operational reliabilities and plasma reproducibility, bookkeeping variation of plasma parameters (cleanliness, frequency of disruptions, etc.), projection (scheduling) vs actual performance
PDX	November 1979	Feedback control (avoidance) of disruptions
PLT	1979-82	Record keeping on apparatus reliability and plasma parameter reproducibility

3.8.7 Stability Limits on Beta

D III	July 1981	7-MW injection-heated D-shaped and doublet plasmas
ISX-B	1979	Circular plasma
	1980	Noncircular plasma
	1981	Noncircular diverted plasma
PDX	September 1980	Beta limit in collisionless with D-shaped cross section
PLT	1979-82	Circular plasma with auxiliary heating
TFTR	1983	Circular plasma with neutral injection
Torus II		Exploration of high-beta tokamak stability region
TERP		Beta limits

3.8.8 Plasma Disruption

Alcator A	June 1980	Disruptions at low values of q ($q < 2.5$)
D III	1979-80	Effect on vacuum wall and poloidal coils
	1979-81	Identification of modes
	1980-81	Modification of j profile with beams
ISX-B	1980	Profile modification with ECH
PDX	November 1979	Feedback control (avoidance) of plasma disruptions
PLT	1979-82	MHD modes, precursor oscil- lations, impurity effects (high Z and low Z), density limits
TFTR	1982	Disruption studies

3.8.9 Equilibrium Position and Shape Control

D III	1979	Doublet equilibrium, shape control
	1979	D equilibrium and shape control
	1980-81	Equilibrium and shape control with 7-MW beam injection
	1981	Shape control by current profile modification with neutral beam heating
ISX-B	1979-81	Noncircular D, ellipse, elonga- tion up to 1.8, finite-beta plasmas
PDX	April 1979	Radial position feedback
	November 1979	Vertical position feedback
	April 1980	Passive stabilization of divertor, $n = \text{zero modes}$
PLT	1979-82	Position control vertical field feedback
TFTR	Early 1982	Plasma-handling studies

3.8.10 Preheating

ISX-B	1979-80	ECRH and UHRH at ≤ 200 kW, 28 and 35 GHz
	1981	ECRH and UHRH at > 200 kW, ~ 35 GHz (?)
PLT	1979-82	100-kHz oscillator, ECRH (?)
TFTR	Early 1982	Breakdown oscillator studies
Tokapole		ECRH plasma startup

3.8.11 Current Buildup

Alcator C	December 1979	High-current-density operation ($I \sim 1$ MA)
D III	1979	In situ pellet startup
ISX-B	1979-80	Together with preheating
	1981	Neutral-beam-driven current
PDX	April 1980	Magnetic limiter and current profile control
PLT	1979-82	Programmed vertical field, interaction with gas injection
TFTR	Early 1982	Current buildup optimization
Macrotron Microtron		Relativistic electron beam injection, heating and current drive
Versator		LH current drive
PRETEXT		Initial stages of tokamak discharge

3.8.12 Heating Scenarios

D III	1980-81	Beam heating during current rise
ISX-B	1979-81	Neutral beam injection and ECRH
PLT	1979-82	Ohmic heating with neutral beam and/or ICRH at various times and powers, interaction of ICRH on beams
TFTR	Late 1982	Heating studies

3.8.13 Fueling Scenarios

Alcator C	December 1979- June 1980	Density buildup in high-n _r plasmas
D III	1979-80	Gas puffing, in situ pellets
ISX-B	1979-80	Gas puffing, single or repetitive pellet injection, beam fueling
PDX	October 1979	Pellet fueling
	September 1980	Beam fueling
PLT	1979-82	Steady and pulsed gasfeed, pellet fueling
TFTR	Mid-1982	Fueling studies

3.8.14 Fusion Ignition

TFTR	1984	Alpha heating studies
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3.8.15 Burn Control and Quench

TFTR	1984	Alpha heating studies
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3.8.16 Current Quench

D III	1979-80	Current decay control
PLT	1979-82	Programming of ohmic heating coil current to avoid dis- ruptions
TFTR	Early 1982	Discharge termination studies

3.8.17 Neutral Beam Heating Physics

D III	1980-81	7-MW near-perpendicular injection in D-shaped and doublet plasmas
ISX-B	1979-80	1.5-MW parallel injection
	1980-81	3.0-MW parallel injection
	1980	Ripple injection
PDX	Spring 1980	6-MW injection

3.8.17 (continued)

PLT	1979-82	≤3 MW co- plus counterinjection, toroidal rotation, beam slowing down, interaction with ICRH, effect on transport
TFTR	Late 1982	20-MW hydrogen
	1983	32-MW deuterium

3.8.18 RF Heating

Alcator A	Completed	LHH coupling and efficiency, low power
	December 1979-80	ICRH antenna characteristics, coupling, heating
Alcator C	1981-82	LHH high-power heating, high-power launching phased array
	1981-83	ICRH high-power heating
D III	1980	Low-frequency (~100 Hz) heating (at separatrix) in doublets
ISX-B	1979-81	ECRH up to 200 kW, 28 GHz and 35 GHz
	1981	ECRH at >200 kW, ~35 GHz
PDX	Late 1981	ICRF or LH if successful on PLT
PLT	1979-82	≤5-MW ICRH at first and second harmonic, mode optimization and tracking interaction with neutral beam, minority species heating, LHH energy deposition to ions and electrons, current drive
Macrotron		ICRF
Microtron		
Versator		LHH, ECRH
Tokapole		ECRH, ICRH
PRETEXT		Alfvén wave heating

3.8.19 Fusion Alpha Heating

TFTR	Mid-1983	First tritium
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3.8.20 Plasma and Impurity Transport in Scrape-off Region

Alcator A	December 1979	Iron, aluminum, molybdenum
D III	1979	Control of boundary parameters by flow reversal and shaping
ISX-B	1981	With and without bundle divertor
PDX	November 1979	Ohmic-heated plasma with poloidal divertor
	August 1980	Beam-heated plasma with poloidal divertor
PLT	1979-82	Surface analysis, low-energy neutral spectrometry, energy- accounting bolometry, limiter heating
TFTR	1982	Limiter optimization

3.8.21 Impurity Generation

Alcator A	1980	Limiter studies, limiter design, carbon limiter, role of wall and limiter in impurity generation
D III	1979	Medium-Z limiter
	1980	Low-Z limiter
	1980	Expanded boundary (D-shaped)
	1979-81	Surface studies
ISX-B	1979-81	Surface coating, erosion studies, arcing studies
PDX	1979-81	Materials configuration and plasma-modification studies
	1979	With OH plasma
	1980	With beam-heated plasma
PLT	1979-82	Surface analysis, neutral beam bombardment, arc tracks, dis- charge cleaning, titanium gettering, limiter materials
TFTR	1982	Limiter optimization

3.8.22 Impurity Removal and Burial

Alcator A	1979-80	Surface physics
Alcator C	1979-81	Surface physics
D III	1979-81	Surface studies
	1979	Flow reversal
ISX-B	1979-81	Surface physics
	1981	Bundle divertor
PDX	1980	MHD rejection and edge modification
PLT	1979-82	Surface analysis
TFTR	1982-83	Surface studies
Macrotor Microtor		Discharge cleaning, titanium gettering

3.8.23 Plasma and Impurity Transport in Plasma-Edge Region

D III	1979	Flow reversal, impurity injection
ISX-B	1979-80	Naturally occurring and externally injected impurities by laser blow-off
PDX	November 1979	Ohmic-heated plasma
	August 1980	Beam-heated plasma
PLT	1979-82	Visible and ultraviolet spectroscopy (carbon, oxygen, iron), fast gas injection
TFTR	1982	Limiter optimization

3.8.24 Plasma-Edge Modification

D III	1979	Impurity flow reversal (IFR)
ISX-B	1979-80	IFR with asymmetric particle source
	1980	Beam-driven IFR
	1980	Edge temperature effects with ECH
	1981	Bundle divertor
PDX	November 1979	Intense gas injection
	August 1980	Edge-turbulence enhancement
PLT	1979-82	Fast gas puffing, low-Z impurity cooling
Macrotron Microtron		Modification of plasma-sheath potential

3.8.25 Wall/Limiter Configurations

Alcator A and C	December 1979- 81	Limiter design
D III	1979	Medium-Z limiter
	1980	Low-Z limiter, neutral beam, armor limiter
ISX-B	1981	With and without bundle divertor
PDX	August 1981	Beam dump and wall limiter
TFTR	1982	Limiter optimization

3.8.26 Poloidal Divertors

PDX	November 1979	Ohmic-heated plasma
	September 1983	Beam-heated, collisionless, high-temperature plasma, optimized divertor configuration
PLT	1979-82	Probably some limited divertor studies
Tokapole		Divertor simulation

3.8.27 Bundle Divertors

ISX-B	1981	Unloading and shielding physics magnetics
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3.8.28 Helium Removal

PDX	August 1980	Plasma jet pump test
QED	Fall 1979	Gas-neutralizer test

3.8.29 Fueling

Alcator A	Completed	Gas puffing (pellet?)
Alcator C	1979-81	Gas puffing (pellet?)
D III	1979	In situ pellet
	1979	Gas puffing
ISX-B	1979-81	Pellet ablation, gas puffing, intense neutral beam heating
PDX		(Covered earlier)
TFTR	1983	Pellet fueling studies

REFERENCE

1. The results of the ORNL TNS Program studies are contained in ORNL/TM-6720-6731, Oak Ridge, Tennessee (1979).

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4. NUCLEAR SYSTEMS R&D NEEDS ASSESSMENT

4.1 NUCLEAR SYSTEMS SUMMARY

Generic R&D must be carried out in several areas in order to directly support the nuclear systems design effort. These R&D requirements are in addition to those plasma physics requirements, such as the characterization of plasma disruptions, that must be met in order to provide a design basis for the various nuclear systems.

The highest priority at present is the generation of experimental data on particle trapping in lithium droplets to establish the feasibility of using a lithium droplet shower as a divertor collector. Such an arrangement has major advantages over alternate concepts and might be applied in the ETF if feasibility can be established. Data to establish such feasibility should be available by the beginning of FY 1981 to allow the inclusion of this concept in the Conceptual Design Phase or to permit the incorporation of an alternate concept.

Data are also needed on first-wall armor concepts and low-Z chamber coating designs in order to reflect feasible design concepts for the Conceptual Design. It is expected that the first wall will have to be designed to withstand a substantial number of plasma disruptions. A systematic experimental program is needed to generate data on candidate materials and design concepts to protect the actively cooled wall from disruptions. Similarly, data on low-Z coating concepts are needed to support the designs of wall concepts that eliminate the sputtering of deleterious material from the chamber walls. These data are needed about six months prior to the completion of the Conceptual Design Phase.

In addition, a program is required to establish the feasibility of utilizing welded and/or bakeable mechanical vacuum seals between adjacent shield sectors. An external welded seal appears to be feasible, and a metallic O-ring seal may be feasible if sufficiently large clamping loads can be applied. This is a major problem, especially if double seals are required. Data to establish the feasibility of these alternatives are required six months before the end of the Conceptual Design Phase.

4.2 INTRODUCTION

ETF nuclear systems consist of the toroidal plasma chamber and the various critical services thereto, including maintenance. Specifically, R&D tasks have been divided into the following areas:

- (1) first wall and first-wall attachments,
- (2) blanket,
- (3) shield and nuclear analysis,
- (4) plasma-chamber vacuum pumping,
- (5) fuel-handling and containment systems,
- (6) remote maintenance, and
- (7) divertor collector.

Although the first wall and first-wall attachments, the blanket, and the shield are all components of the plasma chamber assembly, they are considered separately here because of the unique problems in each area that require investigation and solution. The toroidal chamber is segmented for assembly and disassembly, and several sectors have provisions for one or more experimental blanket modules that will be readily replaceable. Neutral beam ports, or ducts, are provided in several sectors. Vacuum pumping is accomplished through branch ducts from each neutral beam duct. Fuel and tritium handling are accomplished in a system that transfers fuel removed from the plasma chamber, purifies it, separates deuterium and tritium, and stores these isotopes for further use. The divertor collector removes helium ash and chamber-wall impurities from the plasma. The magnet required for this task is discussed in the magnetics system R&D assessment in Sect. 6. Remote maintenance is required after the start of D-T operation for components located inside the reactor shield.

For each major R&D task area specific program requirements are listed in order of priority, with the most critical tasks first. In some areas highest-priority tasks are considered to be those in which the R&D coverage will be adequate if present programs continue on schedule.

The most critical R&D needs in the nuclear systems area are

- (1) particle trapping in divertor collector,
- (2) tritium processing and control [Tritium Systems Test Assembly (TSTA)]
- (3) first-wall armor,
- (4) low-Z plasma-chamber coating,
- (5) torus sector seals,
- (6) remote coolant and vacuum leak detection,
- (7) large compound cryopump, and
- (8) electrical breaks for torus sectors.

Each item is discussed in more detail in one of the seven subdivisions of this section.

4.3 FIRST WALL AND FIRST-WALL ATTACHMENTS

The first wall and first-wall attachments are recognized as critical components in the development of fusion reactors. They are subject to both high 14-MeV neutron wall loading ($\sim 2 \text{ MW/m}^2$) and high surface heat flux ($\sim 0.5 \text{ MW/m}^2$). Also, in order to limit replacement they have to withstand as high a total fluence as is achievable. In addition, these components (or at least their surfaces) must be composed of materials that are not susceptible to sputtering when impacted by plasma particles or that cause minimum impurity effects on the plasma if they are susceptible to sputtering.

Presently the development of the first wall, armor, and limiters is pursued mainly on an ad hoc basis for each experimental device. Some research is being carried out to determine the sputtering characteristics of various candidate materials, and some data are being generated on tritium permeation. In addition, materials data will be generated in the Fusion Irradiation Test Facility (FMIT) program. These R&D activities should be continued and extended to cover materials and conditions anticipated for the ETF program. The additional R&D that must be carried out to support the design of first wall and first-wall attachments for the ETF is indicated in Table 4.1. Figure 4.1 shows a schedule of these activities.

Table 4.1. R&D requirements: first wall and first-wall attachments

Requirement	R&D program status		
	Adequate	Marginal	Not adequate
Develop wall armor concept for plasma chamber			X
Develop effective, reliable low-Z coating for plasma chamber			X
Develop limiter concept, including active cooling			X
Develop cooling method for first-wall panels			X
Generate outgassing rate data for candidate first wall, first-wall armor, and limiter materials as a function of bakeout and discharge cleaning treatment			X

A wall armor concept capable of sustaining neutral beam impingements and plasma disruptions is crucial. These thermal loads are characterized by intense, nonuniform heating on relatively short time scales. As a result, vaporization of the armor material is encountered and high thermal stresses are imposed. At the same time, the high-energy neutron flux damages the material, and both physical and chemical sputtering may erode the surface. Proposed designs should be tested to the extent possible to validate the approach and generate additional data for further design development.

Studies to date indicate that it will be necessary to coat the plasma-chamber wall with carbon or some other low-Z material to minimize the impurity effects of the sputtered material on the plasma. An experimental program in addition to the one at Sandia is required to develop a suitable coating process and to test the adherence of the coating and the sputtering yield.

The development of a suitable limiter concept poses a problem similar to that of the wall armor. In this case the limiter is subject to a longer duration and somewhat more erratic heat load. As a

Fig. 4.1. R&D schedule - first wall and first-wall attachments.

consequence, it may be necessary to actively cool the limiter. This raises the issue of feasibility because local damage due to localized heating may release the coolant, thereby extinguishing the plasma. If this problem cannot be circumvented, provisions will have to be made to rapidly replace a damaged limiter. As with the wall armor, proposed designs should be tested to validate the concept and generate data for any further design development.

In addition, a coolant concept for the first-wall panels must be developed. R&D to support this should be aimed at developing techniques to improve heat transfer into the coolant from the walls of tubes heated on one side. It appears that improvements in heat transfer are needed to prevent damage to first-wall panels during plasma disruptions. Promising techniques should be characterized over a range of temperature, pressure, and heat flux so that the most suitable technique can be selected for the Reference Design.

Further, the outgassing rates will have to be determined for candidate first wall, chamber armor, and limiter materials, including coated surfaces. The outgassing should be measured as a function of time for a range of bakeout conditions. These data are needed to aid in design selection and to provide treatment specifications for the selected design.

The above R&D should be started as soon as budget constraints permit to provide some data for the Preconceptual Design Phase. The bulk of the information should be generated by the end of FY 1982 to provide input to the final Conceptual Design.

RDAC's will be required on the first wall, first-wall armor, and limiter concepts adopted in the Conceptual Design. The objective of these RDAC's will be to validate the design concepts adopted and generate experimental data for detail design. These RDAC's should also be used to develop and validate maintenance procedures and equipment.

4.4 BLANKET

It is assumed that one or more high-temperature, nonbreeding blanket segments will be provided to produce sufficient thermal power for a low-power (~ 5 MW) electric power conversion system (possibly with fission

multiplier). Both helium and pressurized water are considered candidates for cooling the blanket. It may be necessary to protect the blanket with actively cooled first-wall panels or radiatively cooled first-wall armor. The R&D required for those elements is covered in Sect. 4.3.

At the present time there is no R&D specifically oriented toward the blanket for fusion reactors. OFE has requested input on an R&D program oriented toward fusion blankets in general, and any work initiated as a result of this inquiry should be a major contribution to the R&D needs of the ETF. The major R&D needs for the ETF blanket are listed in Table 4.2, and a schedule for these R&D needs appears in Fig. 4.2.

Table 4.2 R&D requirements: blanket

Requirement	R&D program status		
	Adequate	Marginal	Not adequate
Develop mechanical/welded seals for blanket modules			X
Develop structural concepts for blanket modules			X
Develop cooling method for gas-cooled blanket modules			X

A promising concept is the use of radially oriented cylindrical modules for fusion reactor blankets. This concept implies the use of mechanical or welded seals between the modules and the supporting structure. Various seal designs must be developed and tested for ease of maintenance operations as well as seal integrity. This information will then be used for selecting a seal concept and developing further improvements to the selected concept.

Also, structural concepts for blanket modules must be developed and tested. In most designs conceived to date, the plasma-facing wall is highly loaded because of structural constraints. The modules are particularly vulnerable to leaks and/or structural failure due to crack propagation. A comprehensive program integrating design, analysis, inspection, and testing must be developed to ensure the structural integrity of these highly loaded components.

Fig. 4.2. R&D schedule - blanket.

A cooling method that can accept the relatively high neutron loading ($\sim 2 \text{ MW/m}^2$) and, if possible, the high surface loading ($\sim 0.5 \text{ MW/m}^2$) as well also poses a challenging problem. Among the considerations of concern are heat transfer film coefficient, pressure drop, flow stability, and flow stagnation conditions. Proposed designs must be modeled and tested to identify preferred features and validate the designs.

The above cooling and seals R&D tasks should be started as soon as funding is available so that the feasibility of the concepts proposed during the Conceptual Design can be established. Definitive information is required by the end of FY 1983 to provide input to the final Conceptual Design.

RDAC's will be required to support the detail design of the specific blanket module selected for the ETF.

4.5 SHIELD AND NUCLEAR ANALYSIS

The shield system is designed to protect components from nuclear heating, radiation damage, and induced activation and to control personnel radiation exposure outside the shield envelope. Because the shield also forms the plasma chamber, it must be designed to eliminate or stringently limit leakage into the chamber. Provisions must be made to accommodate test modules and to permit rapid disassembly and reassembly for access to interior components.

Improvement in nuclear analysis methods and further development of nuclear data are being carried out in a number of laboratories, including Los Alamos Scientific Laboratory (LASL), ORNL, Lawrence Livermore Laboratories (LLL), Brookhaven National Laboratory (BNL), and Argonne National Laboratory (ANL). Much of the development being carried out as part of the fission program is also applicable to fusion. In addition, some development of codes is directed specifically to the fusion program.

The requirements for the ETF are summarized in Table 4.3. A schedule for these R&D activities is presented in Figs. 4.3 and 4.4.

Assuming the shield sectors are used to form the vacuum chamber, research is required to provide a method to seal adjacent sectors either

Table 4.3. R&D requirements: shield and nuclear analysis

Requirement	R&D program status		
	Adequate	Marginal	Not adequate
Develop mechanical/welded seals for shield sectors			X
Develop electrical break in shield sectors or at joint between sectors			X
Develop outgas cleaning technique for narrow gaps between torus sectors			X
Generate radiation effects data with 14-MeV neutrons for cold-worked austenitic stainless steel, graphite, ceramic materials, super-insulating materials at 4K, and copper (change resistivity at 4K)			X
Investigate effectiveness of copper annealing to offset radiation damage			X
Integrate experiments to explore neutron transport and validate computational methods with more complex shield geometries			X
Generate high-energy-neutron cross-section data for materials under consideration, especially neutron and gamma production cross sections			X
Develop neutron-streaming computer code with improved semiempirical techniques (Monte Carlo with albedos)			X
Develop albedo data for ETF ducts			X
Improve Monte Carlo variance reduction techniques and simplify geometry input			X
Develop improved techniques for routine coupling of discrete ordinate and Monte Carlo codes			X
Improve acceleration techniques for discrete ordinate codes, especially for streaming calculations		X	
Develop more effective methods for dividing large transport problems into subproblems with partial solutions		X	
Complete 3TRAN development		X

Fig. 4.3. R&D schedule - shield.

Fig. 4.4. R&D schedule - nuclear analysis.

by mechanical means or by welding. A method of adequately cleaning the surfaces in these joints will also have to be developed to prevent outgassing problems during operation.

Electrical breaks may have to be provided in the shield assembly to prevent the induction of currents that will inhibit magnetic flux penetration. Because of neutron radiation and outgassing problems, these breaks probably cannot be an epoxy or elastomer material. Ceramic materials, on the other hand, present serious problems in the design of load-bearing structures and vacuum-tight joints. An R&D program is required to evaluate different materials and to develop and evaluate various design concepts. An entirely new material may have to be developed.

Additional high-energy-neutron cross-section data are required for the range of materials under consideration. Additional radiation effects data are required for 14-MeV neutrons on a variety of materials including cold-worked austenitic stainless steel, graphite, ceramic materials, superinsulating materials such as aluminized Mylar, and copper at 4K. Recent data suggest that radiation damage experienced by copper at 4K cannot be completely annealed. Tests on the irradiated copper samples should be performed.

The integral experiments should be extended to include more intricate penetration configurations, such as are encountered with the vacuum pumping ducts, and the actual materials being considered for the ETF shield. One or more engineering mockups may be needed for Title I Design, probably supported by RDAC.

R&D programs should be initiated in these areas as soon as budget constraints permit. Programs are urgently needed to explore potential solutions to the shield joint and electrical break problems because feasibility is an issue for these cases. Although the analysis methods present less urgent concern, work in this area should be aggressively pursued. Radiation effects data should also be aggressively developed because here a feasibility question also exists with respect to organic insulators.

RDAC's will have to be initiated to support the design of the mechanical features of the shield. It is anticipated that design-specific research programs will be required for the shield sector joints and electrical breaks.

Although it might be possible to perform the shielding design of the ETF with existing analytical tools, it should be recognized that further improvements in methods would result in two kinds of cost savings: (1) the development of more efficient methods would save engineering manpower and computer costs; (2) the development of more accurate methods would allow the shield designer to reduce uncertainty factors in shield thicknesses and radiation levels with resultant savings in ETF capital and operating costs. Among the areas in which computer codes should be improved are Monte Carlo variance reduction techniques, albedo methods, and geometry simplification; discrete ordinates acceleration techniques, problem subdivision techniques, and volume coupling methods; and general improvement in the treatment of neutron streaming.

4.6 PLASMA-CHAMBER VACUUM PUMPING

The plasma-chamber vacuum-pumping system is designed to evacuate the plasma chamber at the end of each burn so that a fresh charge of deuterium and tritium can be introduced for each burn. The pumping system is designed to pump down the plasma chamber from a pressure of 10^{-4} torr to 10^{-6} torr in 25 s after each plasma quench. Because of the very high pumping speed requirement, compound cryopumps have generally been specified for this application.

The R&D work on compound cryosorption pumps capable of pumping helium as well as hydrogen species has been limited. The one industrial firm offering to supply cryosorption pumps designed and fabricated the components for a pump for the TSTA but was unable to assemble the pump because of the lack of financial resources. Subsequently, LASL took delivery on the parts and assembled the pump. Some testing data will be developed as part of the program to incorporate this pump into the TSTA.

The R&D requirements for the plasma-chamber vacuum pumping system are indicated in Table 4.4. Figure 4.5 shows a schedule of these activities.

Table 4.4. R&D requirements: vacuum-pumping system

Requirements	R&D program status		
	Adequate	Marginal	Not adequate
Develop compound cryopumps		X	
Develop alternative vacuum pumps			X
Develop large-diameter (~ 1 m) vacuum duct valve			X

The development of a large compound cryopump should be carried out. Design information is needed on the effectiveness of such pumps in pumping helium as well as hydrogen species as a function of arrangement options. In particular, the helium-pumping surface will have to be surrounded by two sets of chevrons, one at liquid nitrogen temperature and one at liquid helium temperature (to pump hydrogen), and it may be desirable to use baffles or reverse surfaces for one set of chevrons. Also, additional research is needed to better identify the cause of momentary loss of pump speed and to develop molecular sieve designs and/or map operating regimes to avoid the phenomenon.

Cryocondensation pumps entail the loading of a substantial quantity of tritium between regenerations. The resulting tritium inventory may present safety problems in the design of the reactor and reactor building. Accordingly, it would be desirable to develop an alternative vacuum pump that does not involve a substantial tritium loading. The most likely candidates to avoid this problem are turbomolecular pumps that do not employ oil lubricants and mercury diffusion pumps with traps to prevent contamination of the plasma with mercury.

R&D should also be carried out on a large-diameter (~ 1 m) duct valve, required to isolate a pump for regeneration while the reactor is operating. Assuming that the most desirable arrangement is a bifurcated

Fig. 4.5. R&D schedule - vacuum-pumping system.

duct that permits one pump to be isolated while a second is on line, a relatively leak-tight valve is required. This requirement, together with the size and environmental considerations, dictates an R&D effort to confirm feasibility and provide basic design data.

Because the type and location of the plasma-chamber vacuum pumps affect other aspects of the ETF Conceptual Design, it is important to initiate the R&D on the compound cryopump and an alternate concept as soon as funds are available. In any case, sufficient data should be made available to make a selection of a vacuum pumping system by no later than the end of FY 1981. The valve R&D is less critical and could be started in early FY 1981 with the confirmatory information available for the end of FY 1982.

Additional research in the form of RDAC's will be required to support the detail design of the vacuum pumping system for the ETF. In particular, information will be required on the performance of large vacuum pumps, and design concepts to accommodate remote maintenance will have to be developed.

4.7 FUEL-HANDLING AND CONTAINMENT SYSTEMS

There are a number of features unique to the ETF that will have a considerable impact upon tritium-handling considerations. For the first time a fusion reactor will have a complete, integrated fuel-recycling system in which substantial quantities (up to 10 kg/year) of tritium will be consumed. The reactor will be surrounded by a large enclosure having a heterogeneity of surfaces that could contain substantial amounts of sorbed tritium. The divertor collector may use liquid lithium and will require tritium recovery rates in excess of those for a full-scale commercial fusion reactor blanket. The first wall and shield will be water cooled (perhaps D_2O), and there will be the potential for excessive ingress of tritium into the coolant. Components exposed to the plasma chamber will experience a significant flux of 14-MeV neutrons along with continuous tritium infiltration. In order to test the viability of various blanket concepts, blanket modules will be tested. These features

require significant advances in the state of tritium technology. Specific areas of concern are listed in Table 4.5, and a schedule for these activities appears in Fig. 4.6.

The fuel-handling system is designed to accept and process the material pumped from the plasma chamber by means of the plasma-chamber vacuum system and from the divertor collector and neutral beam injectors. The system consists of the following subsystems: transfer pumping, fuel cleaning (ash and impurity removal), hydrogen isotope separation, fuel storage, and fuel preparation. A demonstration TSTA is being installed at LA3L for the development, demonstration, and interfacing of technologies related to the D-T fuel cycle for fusion reactor systems, including the ETF. Most of the technical and operational problems related to the fuel-handling system will be addressed in the TSTA. In addition to demonstrating the performance and safety of the overall system, this facility will be used to test and qualify (including reliability testing) components for tritium service.

In addition to the major components of the fuel cycle, there are a number of auxiliary systems such as the emergency detritiation system that will also be tested at the TSTA. However, there are aspects of tritium decontamination procedures that are not currently under study (experimentally) in adequate detail for the requirements of the ETF. These include the problems of tritium sorption and desorption from the complex array of surfaces at various temperatures that could be exposed to tritium, thereby affecting the ability of cleanup systems to achieve effective decontamination.

Because of the large uncertainties in the amounts and the chemical nature of the impurities that will be exhausted from the plasma, it is difficult to design fuel purification and processing systems. Every effort should be made to establish more firmly the levels of all significant impurity species in fusion reactor exhausts, using results from TFTR, D III, and other present-day fusion devices.

The water coolant (perhaps D_2O) will tend to act as a sink for tritium. Permeation and injection of tritium into the first-wall coolant will probably require a barrier of about 100-X effectiveness.

Table 4.5. R&D requirements: fuel-handling and containment systems

Requirement (probable test site ^a)	R&D program status		
	Adequate	Marginal	Not adequate
Demonstrate performance of integrated fuel-handling system (TSTA)	X		
Demonstrate safety of tritium recycle in prototypical tritium-handling system (TSTA)	X		
Develop, test, and qualify components for tritium service (TSTA)	X		
Resolve classification and accountability issues with appropriate personnel in federal government			X
Demonstrate performance of auxiliary safety (cleanup and backup) systems and tritium-monitoring instrumentation (TSTA)	X		
Study dynamics of enclosure cleanup operations following tritium spills (TSTA)		X	
Determine concentration and chemical composition of impurities exhausted from plasma			X
Develop methodology to control tritium in reactor building (TSTA, MF)		X	
Develop tritium recovery unit for divertor			X
Develop test data on tritium sorption/desorption phenomena and tritiated water formation (ORNL, MF, LLL)			X
Characterize tritium interactions (solubility, permeability, etc.) with 316-SS and other ETF materials under realistic conditions (ANL, ORNL, others)		X	
Develop effective barriers to tritium permeation (ANL, ORNL)			
Demonstrate long-term reliability and maintainability of components for tritium service (TSTA)	X		

Table 4.5 (continued)

Requirement (probable test site ^a)	R&D program status		
	Adequate	Marginal	Not adequate
Develop and demonstrate removal of tritium from normal and heavy water (MF, ORNL)	X		
Develop and demonstrate extraction of hydrogen isotopes from liquid lithium (ANL, ORNL)		X	
Quantify liquid lithium handling and safety issues and establish operating plans (HEDL, ANL)	X		
Initiate solid breeder development program (ANL, BNL, others)			X

^aBased upon ongoing funded programs.

Fig. 4.6. R&D schedule — fuel-handling and containment systems.

This barrier may also be needed to minimize H_2 or D_2 permeation into the plasma. The fundamental permeation data for 316-SS are known, and programs are in progress at ANL, ORNL, and elsewhere to develop effective barriers to tritium permeation. The tritium injection mechanism is not as well understood, and the actual tritium ingress rates will depend upon the energy spectrum of plasma-edge particles and thermal conditions at the coolant tube wall. In the event of excessive tritium accumulation in the water, it will be necessary to carry out enrichment. Programs at MF and in Canada are currently developing methods for tritium recovery from normal and heavy water, respectively. It appears that the technology required for controlling tritium in water will be sufficient for the needs of the ETF if heavy water is used as the coolant.

The lithium divertor collector will accumulate ~ 1 kg of tritium per day [two to three times as much tritium as would be bred per day in a 1-GW(e) commercial fusion reactor blanket]. There are ongoing efforts at ANL and elsewhere to develop methods for tritium recovery from liquid lithium. With a reasonable funding profile over the next six to eight years, it should be possible to meet the needs of the ETF in this area. Studies of lithium fires, lithium safety, and related lithium-handling issues are being carried out at Hanford Engineering Development Laboratory (HEDL); that program should provide the input necessary to the ETF.

The ETF will use modular inserts to test various blanket concepts. At present there is only limited information on the behavior of solid breeder materials under anticipated fusion blanket operating conditions. In order to be in a position to design and fabricate solid breeder modules in the ETF, it will be necessary to pursue aggressively a solid breeder materials program.

It is essential to the success of the ETF that the TSTA program be continued aggressively. Because the TSTA is not scheduled for full operation until the end of FY 1981, only part of the required data base will be available for the ETF Conceptual Design. Additional specific design information not covered in the TSTA program or other programs will have to be generated through RDAC's upon initiation of the Title I Design Phase.

4.8 DIVERTOR COLLECTOR

One of the designs for the ETF incorporates a bundle divertor for controlling impurities in the plasma. The bundle divertor reduces plasma/vessel wall interaction and provides a means for collecting the helium ash and impurities sputtered from the vessel wall. It has been estimated¹ that the divertor collector system for the ETF must be capable of pumping 4.6×10^{22} particles/s and absorbing a heat load of 226 MW. Because the target area of diverted plasma is limited by shielding and space requirements, it is further estimated that a heat flux of 10 MW/m^2 must be absorbed. There are presently no known methods for absorbing such high heat flux levels for long periods of time. Therefore, R&D on new concepts for collectors is required for incorporation into a bundle divertor.

It has been proposed that a liquid lithium jet spray at the divertor plasma target area might be capable of satisfying the required ion and heat collection requirements. Liquid lithium at 325°C is an excellent medium for collecting and retaining high-energy ions of hydrogen isotopes. The energy of the ions in the diverted plasma is estimated to be 0.5-3.5 keV.¹ The use of liquid lithium as part of the development of fusion technology has been studied in the areas of blanket cooling. The lithium collector concept could incorporate the existing knowledge for recycling hydrogen isotopes to the plasma. Liquid lithium has also been studied as a first-wall coating over the interior of the torus vessel and as a collector for poloidal divertors. These studies were based on a gravity-driven lithium flow over a metal substrate. The studies show that these concepts are limited by MHD effects and the relatively high vapor pressure of liquid lithium.

It is also suggested that new concepts based on the use of solid collectors be developed. Water-cooled target plates that are changeable without breaking vacuum have been used for the DITE bundle divertor at Culham. Also at Culham a vertically moving target plate is being studied for the Mark II divertor. The collector plate concept being studied at KFA for the TEXTOR bundle is based on a spinning disk with vanes and a water-cooled hub.

The R&D requirements for a divertor collector for the ETF are summarized in Table 4.6, and a schedule is presented in Fig. 4.7.

Table 4.6. R&D requirements: divertor collector

Requirement	R&D program status		
	Adequate	Marginal	Not adequate
Experimentally develop data for probabilities of liquid lithium and solid materials to trap hydrogen ion isotopes, alpha particles, and neutral hydrogen isotopes over a range of temperatures			X
Develop flow model for nozzle design satisfying requirements for droplet size and velocity in varying magnet field		X	
Develop differential pumping methods consistent with vapor pressure and droplet size of liquid lithium		X	
Develop methods of heat removal and optimization nozzle placement to account for varieties of thermal load			X
Develop materials and concepts for solid collectors			X
Develop secondary solid plates to protect housing target chamber during fault conditions			X
Develop methods for testing divertor collector systems			X

Reference 2 points out that the feasibility of the liquid lithium collector system is dependent upon obtaining the following additional data:

- (1) data on the collection of hydrogen isotope ions with liquid lithium beyond the present temperature range of 350°C,
- (2) data on the ability of lithium droplets to retain embedded alpha particles, and

Fig. 4.7. R&D schedule - divertor collector.

- (3) data on the ability of liquid lithium to pump neutral hydrogen isotopes.

The lithium collector concept is dependent upon being able to produce high-velocity droplets that can pass through a nonuniform magnetic field and be collected. The nozzle jet velocity and droplet size must be optimized to limit vaporization. Existing mathematical models and codes can be modified to study the feasibility of using liquid lithium as a collector.

Because the vapor pressure of liquid lithium is 5×10^{-5} torr at 325°C, there is concern that some particles of lithium vapor may condense on the divertor housing while others escape into the vacuum vessel. Therefore, it may be necessary to develop a vacuum pumping system at the flow collector to limit the amount of vapor escaping to the plasma chamber. It will be necessary to develop a pump system capable of maintaining a suitable vacuum while pumping lithium droplets and vapor at ~400°C. In addition, a flow collector should be developed to minimize pumping requirements and the splashing of lithium.

The portion of the plasma being diverted from the inner edge of the scrape-off layer is considerably denser and higher in temperature than plasma diverted from the outer edge. The lithium jet spray, therefore, should be concentrated in the plasma zone with the highest heat load. MHD codes for computing the shape, temperature, and density distribution of the diverted plasma need to be developed.

Also, methods for cooling the collected lithium and for separating the hydrogen isotopes from the lithium must be developed. In addition, methods for recirculating the collected hydrogen isotopes to the plasma chamber need to be developed using the approaches proposed for advanced fuel concepts.

It is mandatory that the testing of newly developed concepts and designs be conducted on subsystem levels. Methods for testing divertor collector systems under high heat flux loads and nonuniform magnetic conditions need to be developed for liquid lithium and solid collectors. Such a program might utilize neutral beam injectors and a vacuum box within a magnetic field containing the collectors.

Because of the present uncertainty regarding the feasibility of the liquid lithium collector, it is proposed that alternate concepts utilizing solid materials be developed. It is suggested that a research program be conducted to select a material that is resistant to hydrogen embrittlement and has good fatigue strength and thermal conduction properties. Design concepts based on a water-cooled solid material capable of moving through the diverted plasma target area should be developed and tested. Concepts for a solid collector can be used as secondary plates lining the target chamber in the event of a failure of the lithium spray collector.

4.9 REMOTE MAINTENANCE

After the start of D-T operation, many reactor maintenance tasks must be accomplished remotely. The capability to maintain the ETF under conditions of high levels of induced radioactivity is not only critical to the machine mission itself but is also critical in demonstrating the availability for follow-on fusion power tokamaks. All radioactive systems and components that will require maintenance or replacement during operational life must be designed for remote maintenance.

Historically the development of remote maintenance equipment has often been initiated after plant equipment design has begun, and the use of existing components has been required because of time and cost limitations. The ETF program offers an opportunity to utilize a more coherent approach. Although many types of commercial remote or robotic equipment are available, few of these devices are adequate for fusion program needs. TFTR equipment experience will provide some input; however, differences in scale, program timing, and design philosophy require that separate programs be provided in support of the ETF. Methodology for reliability and maintainability assessment is well developed and has been applied to commercial fusion power reactor studies as well as to the INTOR study.

The requirements for the ETF are outlined in Table 4.7, and a schedule for activities appears in Fig. 4.8.

Table 4.7. R&D requirements: remote maintenance

Requirements	R&D program status		
	Adequate	Marginal	Not adequate
Develop effective pressure and vacuum leak detection and location methods			X
Develop manipulator-equipped shielded cabs for crane-mounted and floor-supported installations			X
Develop heavy-duty general service servomanipulator with force feedback			X
Develop heavy-duty transport pallet			X
Develop heavy-duty module-handling machine			X
Develop automatic seal-welding devices		X	
Develop remote-welding viewing equipment			X
Develop electrical fault inspection, detection, and damage assessment equipment			X

Leak detection and location and remote nonautomatic repair equipment will be needed to supplement automatic welders. Manipulator-held welding and welding viewing equipment is a critically needed adjunct for spot repairs of unexpected leaks as well as for mechanical and structural repairs. If a satisfactory system is found for TFTR, some input to the ETF may be available.

Most reactor design studies have included manipulator-equipped shielded cabs as an important part of the remote equipment. Both floor-mounted and bridge-mounted versions of this device are necessary because of the complex geometry of the tokamak. A similar vehicle was developed and tested for the U.S. Air Force during the aircraft nuclear propulsion program, but minimal technological progress has been made since that time.

Although light-duty electric servomanipulators are available, much work remains to improve the force-feedback capability and to provide for

Fig. 4.8. R&D schedule — remote maintenance.

less operator effort. The weight capabilities of these devices must be dramatically improved for fusion reactors. Also, some improvement in radiation resistance may be required.

The weight capability of the remote-handling equipment that is presently available is at least a factor of 10 less than that needed for the ETF. Modular shield sectors will require transport mechanisms with precise-motion capabilities. Manipulators and special purpose machines to assist in the movement of large components will require heavy-duty capabilities and variable-force feedback not now available.

Although seal-welding devices being developed for TFTR will provide a starting point, the ETF will require equipment with unique capabilities. Additional automatic vacuum and pressure seal-welding and inspection devices must be available for the wide variety of large and more complex systems and components required by the ETF.

In addition, electrical faults have occurred on almost all experimental fusion devices and can be expected on the ETF. Inspection, fault detection, and damage assessment equipment must be available for those areas requiring remote maintenance.

RDAC's will be required to support the design of remote equipment and features that are an integral part of the ETF design.

REFERENCES

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5. ELECTRICAL SYSTEMS R&D NEEDS ASSESSMENT

5.1 ELECTRICAL SYSTEMS SUMMARY

The major R&D needs for the ETF electrical systems are noted in the physics R&D assessment section (Sect. 3). They are (1) the means of controlling or preventing damage due to disruptions or other major plasma instability, (2) the requirements for startup to minimize the peak OH system voltage, (3) the requirements for a controlled shutdown, (4) the procedure for burn control, and (5) the feedback control for the ETF operations including the above features and plasma stability optimization. Because the R&D needs associated with these features primarily involve plasma physics, they are not reemphasized in this section. The startup procedure and requirements primarily require plasma physics development; however, because of the large impact of the startup procedure on cost, complexity, and reliability, R&D emphasis may be more an engineering need than a physics limitation.

The more direct electrical systems R&D needs are (1) the development of neutral beam systems and an associated test facility, (2) the development of a rf test facility, and (3) the development of a pellet fueling device and its tritium-handling test facility. A priority rating for the electrical systems R&D needs appears in Table 5.1.

The R&D needs for diagnostics associated with reactor-type operations are also important. These include (1) measurements of tritium and deuterium concentrations, (2) measurements of impurity concentrations, (3) measurements of ion temperatures with modest spatial resolutions, (4) blanket instrumentation, and (5) the operation with reliability and ease of maintenance of many diagnostic devices in high-radiation and high-heat environments.

5.2 ASSESSMENT OF R&D REQUIREMENTS FOR DATA ACQUISITION ON ETF

5.2.1 Introduction/Description

Data acquisition as it pertains to a large energy research facility such as the ETF includes all the equipment used to record and measure a

Table 5.1. Priority rating of R&D needs of ETF electrical systems

	Urgent	Important	Desired	May be used	A-Adequate M-Minimal NA-Not adequate	DS-Design study needed EW-Experimental work needed
AC power and energy storage						
Pulsed power lines			X		M	EW
Power conversion						
Verification of rf-assisted startup	X				NA	DS, EW
OH switch development		X			M	EW
Method of preventing damage due to disruption or fast shutdown	X				NA	DS, EW
Neutral beam sources						
High current D^+ >150 keV	X				NA	DS, EW
Director recovery	X				NA	EW
Negative ion (D^-)			X		M	DS, EW
Long pulse >10 s	X				M	EW
High D^+ fraction		X			NA	EW
Neutral shielding	X				NA	DS, EW
Heating					NA	DS
Damage filament, etc.					NA	DS
Zirconium aluminum	X				M	DS
Helium pumping	X				M	DS
Impurity level data	X				NA	DS
Magnetics shielding	X				M	DS
Improved reliability and maintainability		X			NA	DS, EW
Beam test stand	X				NA	EW
Neutral beam power supplies						
Engineering test		X			M	EW
Regulation needs		X			M	EW
Fault energy		X			M	EW
Shielding		X			M	EW
Shutter/bending	X				NA	DS, EW
RF heating						
Wave-launching structures		X			M	EW
Couplers and/or dc breaks	X				NA	EW
RF test facility	X				NA	EW
Megawatt power tubes		X			M	EW
Types of rf heating		X			M	EW
Windows	X				NA	DS, EW
Material testing		X			M	EW
Klystron (long pulse)		X			M	EW
OH startup	X				NA	DS, EW

Table 5.1 (continued)

	Urgent	Important	Desired	May be used	A-Adequate M-Minimal NA-Not adequate	DS-Design study needed EW-Experimental work needed
Process controls						
Insulation radiation damage			X		A	EW
Expanded computer capability			X		A	EW
Improved connecting links			X		A	EW
Diagnostics						
Reliable sensing of super- conducting features		X			A	EW
Measurement of high energy densities		X			A	EW
Reliable sensing operation of devices in high-radiation environment		X			A	DS
Shielding near sensors (prevent streaming damage)		X			A	DS
Standard packaging for remote handling		X			A	DS
Blanket diagnostics		X			A	DS
Window development		X			M	DS, EW
Vacuum seal material development		X			A	DS, EW
Interferometer development		X			A	DS
Measurement of alpha characteristics		X			M	EW
Measurement of ion temperature with moderate spatial resolu- tion	X				NA	EW
Measurement of deuterium and tritium concentrations	X				NA	EW
Measurements of impurity concentrations		X			NA	DS, EW
Data acquisition						
Reliable operation of electronic devices in high-radiation fields		X				DS, EW
Fueling						
Pellet injector		X			M	EW
Operation in tritium environment		X			M	EW

wide variety of physical variables associated with the machine. The data acquisition system includes everything past the transducer that originally produces an electrical signal. Historically many machines have been built with the logic controlling the machine as an entity completely separate from the data acquisition system. On TFTR, however, the data acquisition system is part of a larger system that both controls the tokamak and provides data acquisition services. The latter method is preferred as long as the necessary restrictions inherent in a control system do not interfere with the flexibility and freedom regularly found on data acquisition systems. This last requirement suggests a loose coupling between the control and data systems; the most critical coupling is in the shared data base representing the machine operating conditions.

In the most general terms the data acquisition system for the ETF may be described as a large network of computers and electronic instrumentation devices. Such a system can be divided into two classes of equipment: (1) the general-purpose computer equipment furnished by vendors for a wide market place and (2) the special-purpose equipment developed specifically for the instrumentation of the project. This assessment will deal mainly with the latter equipment and with the communications problems associated with such a computer network.

5.2.2 Present Programs/Status

Computer networks are currently being widely utilized at several fusion energy research installations for both data acquisition and process control. The host or highest-level computers are supplied by vendors complete with operating system software, program development software, network software, and often data base software. These networks are hierarchical, with the highest-level machines managing the data bases and performing data analysis. The lower-level machines are dedicated to specific tasks, such as controlling a particular process or set of acquisition hardware.

Throughout the fusion community the interfacing between the computer network and the instrumentation hardware has been accomplished using the CAMAC standard (IEEE std. 583-1975). Besides the usual advantages

associated with standards such as modularity and interchangeability, the opportunity for instrumentation developers to proceed without regard to a specific computer vendor has greatly improved the availability of the required forms of instrumentation.

The most commonly used forms of data acquisition hardware to date have been the waveform digitizers, devices that consist of an analog-to-digital converter, a clock, and a self-contained memory that, with appropriate control logic, can record a time-varying voltage. These devices have greatly reduced the need for true real-time computing and have also relieved the requirements for enormous bandwidth into the computer. Data are acquired in parallel, then processed and stored serially. The input/output bandwidth of the computers is thus available for process control requirements.

On the opposite side of the computer from the instrumentation front ends, standard computer peripherals have been used for the display and storage of data. Large moving-head disks and high-speed magnetic tape drives are used for storage and retrieval.

Graphic displays associated with computerized data acquisition systems are often special-purpose devices interfaced with CAMAC or other special-purpose hardware because a rather unique requirement exists for displaying large amounts of data nearly concurrently on a large number of display stations. Generally the bandwidth available using relatively low cost terminals interfaced over EIA RS-232 asynchronous lines is too limited for use in this application.

Input for control purposes has generally been accomplished with specially built panels containing buttons and switches. Conventional terminals are used for program development and for controlling data processing tasks.

5.2.3 ETF Needs

The needs of the ETF should be easily met using commercially available computers, peripherals, and instrumentation.

The following advances in technology should make the ETF data acquisition simpler to implement.

- (1) Improved network software will become available as more and more large computers begin to be constructed using network architecture rather than bus structures.
- (2) Microcomputers used in control elements will become more powerful than today's minicomputers.
- (3) Advances in fiberoptic communications links will make the transmission of information from high-radiation zones or other unsafe areas simple and straightforward.

5.2.4 Gaps

The largest gap in data acquisition hardware development is the lack of experience with electronic devices such as microcomputers and waveform digitizer memories operating in high-radiation environments. A better understanding of potential problems should be forthcoming as experience is gained on the newer, "hotter" machines.

A major deficiency of most present-day data acquisition systems is the lack of sufficient on-line storage for data. The current cost of mass storage devices is prohibitive. It is believed that the continued exponential decrease in cost per bit stored should make mass storage economical for the ETF.

5.2.5 Recommended Programs

No particular R&D programs need be undertaken to provide adequate data acquisition facilities on the ETF. The current state of computer technology is adequate to meet today's needs, and the driving force of the business world will continue to push costs down and performance up so that ETF data system designers will undoubtedly have much more power and considerably greater flexibility at their disposal.

5.3 PROCESS CONTROLS

5.3.1 Introduction/Description

The control for ETF startup, burn, shutdown, dwell time, and periods of shutdown is assumed to be similar to present modern facilities utilizing a digital computer system.

Several top-level computers would provide for master control and data keeping, and minicomputers would provide lower-level controls. The interconnection and system for control would be by industrial standards such as CAMAC.

The computer system would be provided with redundancy and uninterruptable power supplies for reliability. In addition to controls, the system would provide monitoring of trouble or faults and information about recommended procedures for correcting these problems.

There would be a close relationship between the control computers and the data acquisition system.

5.3.2 Present Programs/Status

In computer control and data acquisition systems, computer technology is advancing at such a high rate that the equipment available in five or ten years may have little resemblance to that selected for TFTR, JET, or JT-60. There is little doubt, however, that the time scales for the control of the ETF will be easier than for its predecessors and that the hardware capability to handle large amounts of data will be increased.

The evolution in control systems is to continually provide more information and controllability in smaller, less expensive packages. This advancement in the general control field will be adaptable for use on the ETF.

5.3.3 ETF Needs

The ETF control needs that are somewhat different from those of general industry are

- (1) fast response controls for stability or control of disruption and long steady-state burn periods,
- (2) frequent startup and shutdown,
- (3) environmental conditions of high voltage and power, and
- (4) environmental conditions of radiation and high magnetic fields.

5.3.4 Gaps

The major gap associated with the ETF control system and present-day process control systems on fusion experimental devices involves the radiation environment.

It is assumed that there will be a need for more memory capability in the ETF than on present devices but that advancements in the computer field will include increases in memory capability.

5.3.5 Recommended Programs

There are no requirements for controls that cannot be provided with present equipment and components; therefore, there are no recommended R&D programs. When specific control devices are required within the high-radiation area, RDAU's may be required to ensure satisfactory reliability. (The present programs on high-energy-neutron studies on materials and insulators should be continued.)

5.4 DIAGNOSTICS

5.4.1 Introduction/Description

Diagnostics for the ETF is defined as components to detect or sense measurable features and display or make information available. Diagnostics in fusion was originally considered plasma diagnostics. However, the term has been increasingly taking on the more general meaning of metering. In the ETF diagnostics will continue to expand to include this more general meaning. Sensing and displaying measurable factors including radiation, temperatures, and flows within the ETF containment area will

be included as diagnostics. Following are items that are different from those required on earlier fusion experimental devices:

- (1) sensing of superconducting fluid flows and temperatures,
- (2) measurements of high-energy-neutron densities,
- (3) reliable sensing of standard features in high-radiation environments, and
- (4) shielding near sensor to prevent neutron-streaming damage to adjacent equipment.

5.4.2 Present Programs/Status

Plasma diagnostic techniques and related computer data acquisition and processing systems are under intensive development. Existing tokamak diagnostics is being extended, and new measuring techniques are being developed for use on the new generation of larger machines, TFTR, JET, and JT-60. Computer data processing is rapidly evolving as a result of progress in the general area of microcircuit design and fabrication. There can be little doubt, therefore, that many advances in plasma measurement techniques and equipment hardware as well as faster and larger-scale data processing systems will be available by 1984.

In addition to the current development effort in diagnostics, the larger size, higher temperatures, increased radiation flux, and expanded volume of data anticipated for the ETF will require further diagnostic R&D.

5.4.3 ETF Needs

Many current diagnostic techniques will not work under the full-burn neutron flux from the ETF; therefore, the detailed assessment of whether alternative detectors, special analysis techniques, clever radiation shielding, or completely new methods will make the measurement possible must be made.

For diagnostics of the blankets, a key element in the whole experiment, the requirements of the individual types of study are not well known. Because one of the major purposes of the ETF is to provide

testing of blanket concepts, it is clear that rather extensive diagnosis of the blankets should be provided. These data will replace much of the plasma diagnostic information in significance once successful burn operation is achieved. Key blanket diagnostic needs include

- (1) real-time neutron flux detectors distributed throughout the blanket;
- (2) activation foils removable via rabbits to give neutron spectra distributed throughout the blanket;
- (3) thermocouples to give temperature variations during the pulse and to warn of coolant loss or other undesirable happenings, distribution throughout the blanket and coolant;
- (4) pressure monitors and flow meters located at strategic positions in the coolant (water, gas, liquid metal, or molten salt);
- (5) pressure monitors at a few locations to give information on the effects of electromagnetic forces; and
- (6) tritium-breeding rates determined by rabbit extraction of blanket samples or by direct extraction in the cases of liquid lithium and molten salt coolants.

5.4.4 Gaps

The major gap in present programs results from the inability to provide a high-energy-neutron environment.

5.4.5 Recommended Programs

For all the diagnostics a very critical analysis of the environment is required. The engineering monitors have to be examined to determine their position, radiation tolerance, and maintenance and availability needs. The plasma diagnostics system has to be assessed in more detail for accessibility to the plasma, ability to withstand radiation environment, and measurement alternatives. In both cases the configuration of the facility is closely associated with successful measurement, and a close relationship must be developed with the remote maintenance and shielding design.

An attempt to define specific R&D tasks at this early date can only be speculative, and budgets for the assumed tasks are educated guesses. The following paragraphs, therefore, suggest R&D topics as a basis for periodic revision.

Radiation effects

Diagnostic components within the bulk shield and those viewing the plasma will be subject to intense neutron and gamma flux. Windows for visible and infrared spectra are prone to gross damage and luminescent effects. At ANL fused quartz and sapphire windows for TFTR are being studied. The higher flux levels anticipated from the ETF will require additional study on these and other candidate window materials.

Polymers that are attractive vacuum-seal materials will not be applicable to diagnostic equipment within the bulk radiation shield. Even outside the bulk shield the total dose of 10^{10} rad over the life of the machine approaches the limit for polyimide. Hard (metallic) seal valves and flange joints may require special development for large items not developed for previous machines.

Also, the problems presented by transient noise and permanent degradation in detectors must be solved for the ETF environment. The shielding design around the detectors is, of course, an important factor in this regard, but the practicality of massive neutron and gamma shielding to reduce background radiation levels is questionable. It is likely that advances in radiation-tolerant detectors will be required.

Diagnostic development

Spatial and temporal variations in plasma density. The larger plasma dimensions of the ETF and the high ion density range aggravate refraction and attenuation problems in millimeter and infrared interferometers. Development will be required along the following lines:

- (1) choice of wavelength based on plasma frequency, system accuracy requirements, and available equipment;

- (2) selection of system (Mach-Zehnder or Michelson) based on accessibility of antenna, refraction, attenuation, etc.;
- (3) antenna design;
- (4) source development to achieve power and stability at selected wavelength; and
- (5) signal processing.

Alpha particle distribution and time evolution. A technique for making these important observations has been suggested by D. Post of PPPL. A basic charge exchange experiment is used to create neutralized alphas that escape the plasma into a charge exchange detector. The incident neutral beam is likely to be 800-keV hydrogen. This high-energy beam will require development of a suitable negative ion source and beam line. This method presents a major problem with the large, dense ETF plasma. The neutral alpha must be able to leave the plasma to reach the detector, although the mean free path is shorter than the plasma dimensions. The attenuation of neutrals must be compensated by a sufficient flux from the incident beam.

Other suggested diagnostic developments include

- (1) measurement of fluctuations by millimeter on far-infrared scattering,
- (2) electron temperature by heterodyne microwave detector,
- (3) neutron energy analysis,
- (4) internal poloidal magnetic field measurement, and
- (5) thermal cycling of vacuum vessel interfacing components and development of temperature measurement of materials surfaces inside the vacuum vessel.

When the port access, shielding, and procedure for remote maintenance are determined, there will be a need for RDAC's to reconfigure some diagnostic sensors.

5.5 RF HEATING

5.5.1 Introduction/Description

There are two functions of rf heating. The first is to assist the OH system in initiating the plasma. This provides energy to overcome

radiation losses and greatly reduces the size of the OH system. The maximum OH voltage is reduced from several hundred volts per turn to ~ 10 V/turn.

The second function of rf heating is to heat the hot plasma to the final stages of ignition. In the ETF rf heating could augment the neutral beam system or possibly replace it. One method of augmenting the neutral beam system is to provide heating in the central section, which could not be heated by 150-kV neutral beam systems.

5.5.2 Present Programs/Status

Ion cyclotron resonance frequency (ICRF) heating

Because ICRF heating was first introduced into tokamaks in 1973, experiments at power levels up to ~ 200 kW have been conducted on the ST, ATC, PLT, the Tokamak Fontenay-aux-Roses (TFR), T-4, and TM-1 Vch tokamaks. The present experimental status is summarized in Table 5.2. There are plans to carry out megawatt-level experiments on the PLT, Alcator, TFR, and T-10 tokamaks. The main purpose of these high-power experiments is to determine the feasibility of employing ICRF heating in hot, reactor-sized plasmas. Consequently, the emphasis of these studies is on evaluating the physics of the wave/plasma interaction to permit extrapolation to reactor conditions.

Once the physics base is established, the effort will shift to the development of wave-launching structures suitable for the reactor environment.

The goals of the larger planned experiments are given in Table 5.3. These plans are aimed at producing significant heating in the hottest plasmas available to assess the physics of the wave damping, the damping profile and the means to control it, and the potential of the several heating schemes for extrapolation to reactor conditions. As indicated by the initial results on PLT and TFR, the physics of ICRF heating and its relevance for reactor conditions should become clear during the next few years from the experiments performed on the larger, higher-current tokamaks. With the recommended development program these results will

Table 5.2. Experimental status of ICRF

Device	f (MHz)	P _{rf} (kW)	Launching structure	Coupling efficiency (%)	Heating efficiency (%)	ΔT_i (eV)	Pulse length (ms)	Heating mechanism
ST	25	<450	Two 1/2-turn coils	90	20	200 D ⁺	1	Fundamental two- ion hybrid
ATC	25	<200	1/3-turn copper strip	90	<40	200 D ⁺	10	Two-ion hybrid
T-4	23	50	3/4-turn coil			200 D ⁺	1	Direct minority heating
TFR	50-90	80	Tandem 1/2-turn coils	80	70	150 D ⁺	30	Two-ion hybrid
PLT	25	200	1/2-turn coil	70	>70	250 (up to 60 keV)	70	Two-ion hybrid, D ⁺ (direct minority heat- ing, H ⁺)
TM-1 Vch	7-35	150	Two asymmetric coils	80-90	30-40	100	0.8	Two-ion hybrid
DIVA	50	<100	1/2-turn coil	65	20	100 H ⁺	1	Second harmonic
Macrotron	1-20	≤200	Rectangular loop			200	2-4	Off resonant
TO-1	48	<5	Coil			50	20	

Table 5.3. Planned large tokamak experiments

Device	f (MHz)	Expected P_{rf}	Launching structure	Pulse length	Heating mechanism
Macrotron	1-20	1 MW	Rectangular loops	10	?
Alcator A	150-200	200 kW	Tandem coils	10	Second harmonic
Alcator C	150-200	>4 MW	Tandem coils	10	Second harmonic
PLT	25	5 MW	Multiple 1/2-turn coils	200	Two-ion regime
	55			200	Second harmonic
TFR	60	1 MW	Tandem coils	10	Two-ion regime
T-10	60	1 MW		100	Two-ion regime

be available in time to impact the choice of primary heating technique for the ETF.

Lower hybrid heating (LHH)

The phased waveguide arrays are the most desirable for reactor application and offer the potential for tracking the parallel wavelength if required. These waveguide array couplers are adequately described by a theory developed by Brambilla; quantitative agreement with the basic features of this theory was first demonstrated on the linear device H-1 at PPPL. These results have been subsequently substantiated in tokamak experiments. Concern remains about high-power nonlinear effects on coupling.

Recent CO₂ laser-scattering experiments in Alcator A with a two-waveguide array launcher have detected the lower hybrid (LH) waves inside the plasma, but they have slower phase velocity than expected.

Ion heating. One physical process for transferring the LH wave energy to the plasma ions is ion Landau damping of either the incident slow LH wave or the warm plasma waves produced by linear mode conversion. This heating mechanism was most clearly demonstrated by ion heating in the Vega and JFT-2 experiments. A competing heating mechanism is the damping of short wavelength products of parametric decays. Evidence of this heating was observed in ATC and JFT-2, although the JFT-2 experiments indicate a saturation of the heating due to the decay waves.

Landau damping. By choosing the frequency of the LH wave large enough to avoid linear mode conversion inside the plasma, the energy can be transferred preferentially to the electrons by electron Landau damping. Strong quasi-linear Landau absorption of the wave energy by electrons also requires that the parallel wavelength be chosen to satisfy $\omega/k_{\parallel}v_e \sim 3$. A test of this scheme on toroidal devices is presently in progress on both the D IIA and Octopole devices at GA.

The present U.S. program plan includes two major LHH experiments, on PLT and Alcator C, that will hopefully provide the required demonstration of significant heating along with a confirmation of the physics basis necessary before LHH can be a viable choice for the ETF. A test

of LH current drive is also planned on PLT and Versator II. At present no approved plans exist for a follow-on to these experiments.

There is no test information available on rf-assisted startup. The initial testing was conducted on ISX in FY 1979.

Heating methods other than neutral beam and rf heating

In addition to neutral beam injection and the various rf techniques discussed in the preceding sections, a number of other methods for heating tokamak plasmas have been proposed.

Of all the other heating methods that have been proposed, only one — adiabatic compression — has been demonstrated to work at all in a tokamak of significant size with significant energy confinement times. However, compression appears to have practical benefit only when employed in conjunction with some other non-OH methods.

At this time it seems likely that relativistic electron beam (REB) injection will be tested in one or more small tokamaks, mainly to investigate its potential for current driving. Turbulent-heating experiments will proceed in small toruses outside the U.S. However, major-radius compression is the only other heating method that is planned for use in TFTR, JET, or JT-60 or that could be confidently scaled to the ETF.

5.5.3 ETF Needs

An ICRF heating system for startup and ignition of the ETF consists of a rf source, power transmission and matching system, and the wave-launching structure or antenna. Some relatively minor extensions of the present technology of the power source and transmission system will be required for high-power, long-pulse [or continuous wave (cw)] operation. However, a concerted R&D effort will be required to evaluate the possible launching structures suitable for reactor operation under conditions that will permit optimization of the antenna design. A startup modeling and reactor design study has been conducted for ICRF heating for the New University of Wisconsin Tokamak (NUWMAK) conceptual reactor, and a

conceptual ICRF system and antenna designs have been considered for the Experimental Power Reactor (EPR).

External component technology, which embraces all components outside the neutron shield (i.e., excluding the wave-launching structure and its power feed), is well in hand but will benefit from some modest improvements. Megawatt tetrodes and coaxitrons suitable for ICRF heating sources currently exist with efficiencies of $\sim 90\%$ but are now rated for pulse lengths of only 0.1-0.35 s at peak power. Although cw operation is now possible at the megawatt level for a single tube, development may be desirable to obtain the long-pulse, 5-100-s (essentially cw) operation envisioned for the ETF at the highest possible power per tube. Power transmission can be accomplished easily with 10-20-MW capacity, 23-cm-diam lines that are commercially available. Furthermore, in the ICRF frequency range, transmission over hundreds of feet incurs negligible loss of power, permitting the source to be located external to the reactor containment building and thus removing the need for remote-handling maintenance of the source. Although radiative cooling of the line may suffice, liquid cooling with, for example, borated water should be investigated. Finally, double-stub tuners are presently used to match the antenna feed impedance to the source and are likely to work well with minimal modification for the ETF device. However, an optimal scheme should be developed during the next decade.

The one area for which the technology is only crudely developed is that of launching structures and feeds suitable for reactor operation. To date work has concentrated on establishing the physics base with structures and feeds already proven reliable for current devices. In all previous ICRF experiments coupling loops have been employed to launch the waves; this same technique will also be employed in experiments planned for the near future.

Efficient wave coupling $>90\%$ is easily achieved with loop-antenna systems located in the shadow of the plasma limiter when such coupling systems are suitable. This includes all devices for which the heat load does not seriously increase the resistance of the loop elements. It is possible that loop systems can be developed for the ETF with adequate

cooling of the system elements. The all-metal antennas recently employed on TFR and T-4 suggest that such cooling may be feasible. (All elements, including shields, are of sufficient size to permit liquid cooling.) Furthermore, insulators within the radiation shield may be eliminated, thereby avoiding unnecessary neutron damage of the loop system.

In addition to loop coupling, other wave-launching structures need to be seriously explored for the reactor regimes. Flush-mounted cavity apertures, dielectric-filled waveguide, ridged waveguide, and strip-line couplers are possible contenders. As the machine size is scaled up, these structures become more attractive, although remaining virtually unexplored experimentally.

The feeds to the launching structure in the ETF can also be similar to the coaxial transmission lines now in use. These can be routed around magnetic field coils and at angles through the radiation shield to minimize neutron flux escape. Placement of the vacuum break will be crucial to avoid excessive neutron damage.

Power levels through ports on ICRF experiments have ranged from 1-10 kW/cm² without breakdown, and peak electric field intensities of 10 kV/cm have been obtained in the loop region. It is expected that these levels will remain at comparable values as scaling to the ETF is achieved, depending on the success of the R&D effort envisioned.

In the area of technology, LHH appears to be the most attractive of all present heating options. The existing technology is capable of providing the required power in the appropriate frequency range with high conversion efficiency. The ability to remotely site the power supplies, transport power large distances with waveguides through convenient paths with bends, and the simple, potentially radiation resistant coupling structure provided by waveguide arrays considerably simplify the engineering requirements on the auxiliary heating system. The engineering design of a complete LHH system provided an estimated overall dc-to-rf efficiency of 61% without thermal recovery.

The major part of the waveguide run from the klystron to the launcher will be pressurized with SF₆ to avoid breakdown, and the windows separating the pressurized and evacuated regions should be removed as much as possible

from the direct neutron flux. A dc break is required in the feed waveguide to prevent possible ground-loop interference and ensure personnel safety. It may be necessary to have a pressurized waveguide section through the point of electron cyclotron resonance. Waveguide constructed of beryllium-coated copper appears to be the desired choice of waveguide material directly exposed to the plasma. Window material will probably be BeO or Al_2O_3 , unless radiation resistance is the most important consideration; in that case Y_2O_3 would be preferred. The waveguides and grills should operate at low temperatures to avoid radiation-induced swelling; hence, areas exposed to considerable neutron irradiation will require cooling.

The waveguide array grills have been tested with power densities up to 5 kW/cm^2 without breakdown, and the planned Alcator C experiment is designed for 10 kW/cm^2 . Nonlinear effects, including parametric decays and pondermotive effects, that can modify the wave coupling and propagation may restrict power densities to values lower than the technological limits. Coupling efficiencies $>90\%$ have been achieved with a four-waveguide array.

For initiating the plasma, electron cyclotron resonance heating (ECRH) is most interesting. The injected power per unit port area should be very high because high-frequency gyrotrons are inherently high-power-density devices. It may be possible to use optical waveguides to conduct the gyrotron radiation to the reactor environment. In addition, the use of mirrors will facilitate access and prevent neutron streaming into the gyrotrons. The positioning of a polarizer that transforms circularly polarized gyrotron radiation to polarized radiation could be a major problem.

The main obstacle to the application of ECRH is the present primitive level of tube technology at $f \gtrsim 35 \text{ GHz}$. High-frequency gyrotron tubes with power levels on the order of 100 kW and efficiencies $\gtrsim 20\%$ must be developed by the mid- to late 1980's.

5.5.4 Gaps

A more concerted effort is needed to obtain data on rf-assisted initiation of the plasma such that the number of OH system volt-seconds and the maximum voltage can be kept low.

The heating test program on the current generation of tokamaks should continue until the prospects for ICRF heating are fully established. Subsequent heating experiments for the attractive heating modes should then be performed on the next generation of tokamaks (e.g., TFTR and JET), where excessive heat loads and neutron deterioration of performance will have to be avoided. The next step would be the application of ICRF heating to an ignition in the ETF.

In order to ensure the timely development of the components needed for the ETF, a Radio Frequency Test Facility (RFTF) should be established as soon as the physics base warrants, probably in only one or two years.

The optimization of the loop-coupling structures on PLT, TFTR, etc., will serve to delineate the degree of wave spectrum selectivity required. This information can then guide the design of new types of coupling structures for testing on suitable devices, preferably on RFTF. In this manner all of the development required to optimize operation on TFTR, and subsequently on the ETF, can be done off-site and can lead to highly reliable operation on the ETF.

In addition to testing structures in the plasma environment, materials testing relevant to ICRF heating is needed. RF conductivity and the effects of surface blistering should be measured for candidate conductors (copper, vanadium, molybdenum) at neutron and plasma flux levels comparable to those expected in the ETF. Although it is expected that all insulators (sintered beryllia, alumina, yttria) would be located out of direct view of the neutron flux, their compatibility with conductors, thermal stress, and neutron bombardment should be studied.

Finally, optimization of the external components should be pursued. In particular, 1-5-MW sources in the frequency range 25-75 MHz that are capable of long-pulse operation (practically cw) should be developed.

Clearly the pressing need for the LHH method is a definitive demonstration of significant heating in a tokamak and a corollary confidence in the physics understanding. The planned experiments for PLT and

Alcator C are aimed at this purpose. It is also essential to determine in these experiments any nonlinear limitations on coupling and heating to ensure the optimum choice of the required frequency, parallel wavelength, and port area for the ETF.

The primary additional need for LHH is the demonstration of long-pulse, high-power operation, perhaps on TFTR or D III, to determine if impurity influx is significant and to provide a test of the engineering reliability of the complete system. An additional need exists for engineering optimization of coupling structures, tests of phase tracking, and studies of nonlinear effects; these tests can be done more efficiently and economically on a smaller dedicated rf facility.

Tests of the other potential applications of LH waves such as current drive and current profile control would be desirable.

The selection of LHH as the primary auxiliary heating system for the ETF would make modest development work on the klystron sources beneficial. The development of long-pulse klystrons of 1 MW or larger and improved efficiency are the key objectives. The need for system optimization, reliability studies, and material selection will continue.

5.5.5 Recommended Programs

For ICRH, LHH, and ECRH, a concerted R&D effort is recommended to evaluate the possible launching structures and to obtain high-power, long-pulse operation (>5 s). A RFTF should be established and continue until the prospects of rf heating are fully established for use on the ETF. For ECRH, higher-power and higher-frequency gyrotron tube development should continue.

Although neutral beam heating is now seen as the primary heating source for the ETF, there is a need to determine the effectiveness of rf heating for future devices. The present programs should be aimed at proving the feasibility of rf heating and the single best rf system that could be used on the ETF. Putting a single rf system on the ETF to determine its effectiveness and reliability for future devices is desirable, but trying all possible systems, if a single leading candidate can be determined in a less expensive RFTF, is not recommended.

The most urgent recommendation is to provide proof of using rf-assisted plasma initiation. If it can be shown experimentally that rf heating is an effective method of heating the plasma over the radiation boundary, then the cost and complexity of the ETF can be kept low. However, if high induced OH voltages are required, the coils inside the TF system will require more space and energy.

5.6 NEUTRAL BEAM INJECTORS

5.6.1 Introduction/Description

At the present neutral beam injection offers the best assurance of providing adequate heating for the ETF plasma. The effectiveness of high-power neutral beam injection heating has now been demonstrated in several tokamaks.

Heating the ETF plasma to ignition conditions seems feasible using neutral beam energies achievable with positive-ion-based technology (120-200 keV per deuteron), but with significant advancements beyond the TFTR beam technology required to obtain long pulses at high duty factor, as well as high currents at the upper end of this energy range. Other required developments are an increase in overall efficiency to be obtained primarily by direct energy recovery and increased reliability and component lifetime.

Potentially serious disadvantages of the beam heating method are the exposure of the injector internals to the fusion neutron flux and the expected high capital cost of beam systems. The difficulties incurred by the direct exposure of injector components to neutrons can probably be overcome, especially if the beam injectors can be closed off after startup.

5.6.2 Present Programs/Status

Table 5.4 summarizes the beam heating experiments planned for U.S. tokamaks in the next four years. The results of these experiments should provide the more detailed data base necessary for the confident application of neutral beam injection at longer pulse lengths.

Table 5.4. Planned neutral beam injection experiments on U.S. tokamaks

Device	Year	Beam power (MW)	Beam energy (keV)	Pulse length (ms)
PLT	1979-80	3.2	40	200
ISX-B	1979-80	1.8	40	100
		3.0	40	100
PDX	1980-81	6.0	50	500
D III	1980-81	7.0	88	500
	1982-84	20	80	1000
TFTR	1982-83	35	120	500
	1983-84	35	120	1500

Table 5.5 gives the state-of-the-art parameters of neutral beam injectors. On operating tokamaks the PLT/ISX neutral beam systems have achieved the following parameters: $E_b = 45$ keV, $t_b \leq 0.4$ s, $P_b = 600$ -kW injected H^0 per beam line with overall power efficiency $\eta_I = 0.25$ - 0.35 , and $P_b = 850$ -kW injected D^0 per beam line with $\eta_I = 0.45$. The effective transmission efficiency $\eta_t \sim 0.55$ ($\eta_t = \eta_I$ /neutralization efficiency). The six-source 2XIIB injectors have achieved these parameters: $E_b = 18$ keV, $t_b = 10$ ms, and $P_b = 3.5$ -MW injected D^0 per beam line with $\eta_I = 0.42$ and $\eta_t \sim 0.50$.

In the present beam development program, Lawrence Berkeley Laboratory (LBL) and ORNL have operated ion sources at 110-120 kV and 15-20 A for $t_b \geq 0.25$ s. The TFTR development source has operated at 120 kV and 65 A up to the power supply limit of 20 ms. The TFTR prototype neutral beam system, which is just beginning operation at LBL, will have one of three sources installed and powered by a 120-kV, 65-A, 0.5-s power supply. The projected η_I is 0.24 for 120-keV D^0 and η_I is 0.37 including fractional-energy components. By the end of 1982 three of these beam lines should be operating on TFTR, each delivering 6.5 MW of 120-keV D^0 and 8.4 MW including all D^0 components.

Operating and planned beam lines on tokamaks do not yet include direct recovery systems. The results obtained on test stands are shown

Table 5.5 Neutral beam injector technology based on positive ions

	Present status		Expected status in 1983 on fusion machines	ETF requirement
	On fusion machines	On test stands		
Ion source				
Current density (A/cm ²)	0.4 at 40 keV	0.4 at 40 keV 0.3 at 120 keV	0.3 at 120 keV	<i>a</i>
Extracted current (A)	65 at 40 keV	100 at 50 keV 65 at 120 keV	80 at 80 keV 65 at 120 keV	<i>a</i>
Monatomic fraction	0.85	0.85	0.90	≥0.90
Gas efficiency	0.50	0.50	0.50	0.50
Maximum pulse length (s)	0.4 at 30 keV	1.5 at 120 keV 10 at 25 keV	1.5	5-100
Lifetime				
Source (s)	10 ⁴ -10 ⁵	10 ⁴ -10 ⁵	10 ⁵	3 × 10 ⁵
Accelerator (s)	>10 ⁵	>10 ⁵	>10 ⁵	3 × 10 ⁵ (in neutron field)
Beam line				
Beam energy (keV)	45	120	120	150-175
Neutral beam current ^b (amp-equivalent D ⁰)	30 at 40 keV 250 at 20 keV	40 at 50 keV	90 at 120 keV 190 at 80 keV	200 at 150 keV ^c
Pumping speed (liters/s)	4 × 10 ⁵	3 × 10 ⁶	3 × 10 ⁶	1 × 10 ^{7c}
Transmission efficiency	0.55	0.75	0.75	~0.8
Overall power efficiency ^b	0.45 at 45 keV		0.35 at 120 keV	0.40 at 150 keV
Direct recovery efficiency	0	See Table 5.6	0	≥0.6
Power density at torus port (kW/cm ²)	1.5	2.0	4.0	<i>a</i>

^aTo be determined.^bIncluding full- and fractional-energy beam neutrals.^cAssuming four beam lines to supply 75 MW of full-energy power.

in Table 5.6. So far there has been no demonstration of direct recovery with the combination of beam energies, beam power densities, and pulse length characteristics of existing or future injector systems.

5.6.3 ETF Needs

Long-pulse beam development

Although the reference pulse length for heating the ETF plasma to ignition is 8 s, it is highly desirable to have a pulse length capability of 100 s (the duration of the burn). This requirement implies essentially steady-state operation for the duty factor of 0.8; however, in practice only a fraction of the heating capacity is likely to be needed for an extended period throughout the burn. On present-day test stands, ORNL has operated a PDX ion source at 40 kV and 100 A for 0.5 s and a relatively small ion source at 25 kV and 8 A for 10 s. LBL has operated a fractional-area TFTR ion source at 120 kV and 8 A for 1.5 s. Neutral beam systems operating by 1983 on PDX, D III, TFTR, and MFTF are expected to have a 1-2-s pulse capability. Although no insuperable problems in extending the pulse length as required are expected, a major development program is needed.

Direct energy recovery systems

Figure 5.1 shows projections of η_I vs E_b for ETF-like beam systems with various direct recovery efficiencies. This figure is the result of an analysis that accounts for losses due to beam optics, fractional-energy components, reionization of beam neutrals, less-than-equilibrium density in the neutralization cell, and charge exchange, ionization, and secondary-electron production in the grid gaps.

Direct recovery is desirable for reducing the power supply capacity (and cost) as well as for reducing power-handling requirements in the beam line. Higher efficiency obtained with direct recovery is also a valuable step toward commercial reactor requirements.

Table 5.6 Direct energy recovery experiments

Laboratory	Beam energy (keV)	Pulse length (s)	Total current ^a (A)	Total current density (mA/cm ²)	Ion current (A)	Ion current density (mA/cm ²)	Recovery efficiency
LLL	100	0.5	2	7	1.5	5	0.6
ORNL	37	0.1	20	200	7	70	0.6 ± 0.2
Fontenay	70	1.0	20	250	13	150	0.45
Needed for ETF	150-175	5-100	≥20	>250	≥13	>150	≥0.6

^aIons plus neutrals.

Fig. 5.1: Overall power efficiency, defined as neutral beam power delivered into the torus divided by total electrical power into the injector.

Effects of neutron environment

There are no experimental data and little analysis on problems resulting from neutron streaming into neutral beam systems. There will be no on-line damage data by 1983, but it has been speculated that there may be degradation in the performance of filaments, grids, and insulators. Figure 5.2 shows the falloff in neutron flux, including backscattered neutrons, with distance up to the beam line. Although a definitive analysis needs to be made, a comparison of the streaming neutron flux with the damage thresholds for various materials indicates that damage effects may be tolerable if the ion source and accelerator structure can be replaced semiannually, for example. Heating of cryopanel panels may be so significant that the use of cryosorption pumps or getter pumping should be examined carefully.

If radiation damage of the ion source and/or accelerator structure is a problem, a potential solution is a beam-bending magnet to allow the source to be out of direct line of sight from the plasma. Other possible advantages of a bending magnet include the reduction of (1) streaming gas and (2) lower energy beam components going to the plasma. However, because a magnet introduces problems with beam optics and control that can more than offset any advantages, it should be avoided if possible.

A more likely solution to the problem of damage caused by neutron streaming is a 30-cm-thick plug valve that is closed after ignition is attained. The use of the plug valve would also result in a large reduction in the heat load on the cryopumping system. This bulky valve would have to be opened and closed reliably at least once every cycle, probably by rotation. However, the valve could not be used with any injectors that might have to operate for a substantial fraction of the burn pulse.

In any event it appears that the lifetime of ion source components will be limited by the present-day problems of erosion, thermal cycling, and so forth, rather than by neutron bombardment.

Fig. 5.2. Neutron flux, including backscattered neutrons along the wall of a duct perpendicular to the vacuum vessel, for a fusion neutral wall loading of 1 MW/m².

Beam line pumping system

A PLT beam line cryopump has a speed of 4×10^5 liters/s and requires 2.5 h for regeneration. The TFTR cryopump has a design speed of 3×10^6 liters/s and a regeneration time of 8 h. Although the TFTR pumps could operate safely for 10,000 1-s shots from the viewpoint of hydrogen-explosion, the need to limit tritium accumulation results in more frequent regeneration.

For the ETF beam lines cryopumping could provide sufficient pumping speed and acceptable regeneration intervals, but it has two potential disadvantages. Buildup of tritium inventory may require unacceptably frequent regeneration, and cryopanel heating by neutrons may be significant. Zirconium-aluminum getter pumping should be investigated as an alternative because it would not release the tritium in the event of a vacuum leak and is not sensitive to small temperature rises.

If a significant amount of helium must be pumped into the beam lines, special cryosorption or cryotrapping pumps may have to be developed.

Species mix

Figure 5.3 shows the fraction of neutral beam power at full energy for the ETF probable energy range of $E_b = 120\text{--}200$ keV. The monatomic ion fraction f_{+}^D of the extracted ion beam is as large as 0.80–0.85 in operating systems and test stands. A reasonable projection for most systems operating in 1983 is $f_{+}^D = 0.90$. This value or higher is desirable to ensure adequate plasma^D penetration by most of the neutral beam and to maximize the usefulness of direct energy recovery systems.

Deliverable power density

The PLT neutral beam system delivers 1.5 kW per cm^2 of torus port area. By 1983 the TFTR beam system is expected to be delivering 3 kW/ cm^2 while the D III system delivers about 4 kW/ cm^2 at the torus.

For the ETF the power per unit area should be maximized in order to minimize the size of the injection ports, thereby minimizing the effects

Fig. 5.3. Fraction of neutral beam power in the full-energy component for various species mix in the ion beam.

of neutron streaming. A high power density is more difficult to achieve with the long duct necessary to penetrate the ETF blanket and shield. Because of the ambient magnetic field, beam neutrals that are ionized by collisions with gas molecules in the beam duct are deflected to the duct walls. Bombardment of the wall may liberate more gas, leading to an enhanced rate of beam ionization and a partial blocking of the beam. The duct pressure buildup observed in the PLT experiments must be further understood because this phenomenon may impose design limits on power density and duct length.

The potential for significant duct reionization losses can be reduced markedly by designing high-conductance ducts, insofar as such design is compatible with the need to minimize neutron streaming effects, and also by ensuring a low gas evolution rate from the duct wall through the proper choice of wall materials and coatings. It is important that neutral beam test facilities have the capability to simulate tokamak magnetic fields in the duct region so that the extent of reionization losses and the effectiveness of countermeasures can be determined.

Determining the appropriate balance of the power densities per beam line with the number of ports on the ETF will require a number of specific neutral beam designs and total layouts. More extensive prototype testing than is part of current development practice will be necessary to establish relationships between power density and reliability/availability.

Availability, maintenance, and reliability

Availability. The requirement that the ETF have a high availability (up to 50% is the present goal) is a stringent requirement on systems (such as the neutral beam system) that require routine maintenance. With a startup time of 5-10 s and as many as one pulse every 120 s, each injector will be required to operate for up to 10^5 s/month. (Those injectors with longer on-times during each pulse would have a correspondingly greater requirement.)

Maintenance. Simple hot filaments used as cathodes generally have a lifetime of 10^4 - 10^5 s, and it may well be necessary to replace even

more advanced cathodes and other components subject to erosion (such as the grids) on a monthly basis. Neutron-irradiation experiments are required to determine the effect (if any) of neutron bombardment on normal replacement intervals.

Reliability. Each component must have very high reliability in producing the rated power during the rated lifetime. For example, if there were 20 ion sources, each with a 99.5% probability of successful operation on each pulse, then on the average the system would provide 90% of the maximum rated capability.

Redundancy. It is important that components requiring regular maintenance (e.g., the ion source and recovery system) can be isolated so that the rest of the system can continue to operate. If 60 MW of power were required and produced by four modules, each with three injectors all working perfectly, it would be advisable to have an extra module because module failure would lead to an extended downtime. In addition, to permit regular maintenance during operation, it would be desirable to have an extra injector per module. Thus, we arrive at five modules, each with four injectors, rated for 100 MW total. To give 60 MW there must be four modules always operating (rated for 80 MW maximum) and requiring 98% reliability of each of the 12 injectors.

Power supplies

By 1983 several additional high-voltage power systems should be operational: 120 kV, 65 A for TFTR and 80 kV, 80 A for D III and MFTF. Because of long lead times these systems were specified at a time when little or no operational experience with very high-voltage ion sources was available. Thus, in many respects these systems are similar to those used on test stands, with good regulation, a wide range of voltage selection, and the capability for frequent interrupts on all supplies. For the reactor-like conditions of the ETF the sources will have to have evolved to a higher level of reliability, and certain simplifications that may be possible in the power system should be investigated: the elimination of high-voltage regulation by feedback regulation of the arc power supply to maintain an optimally focused beam at various accelerator

voltages; no fast high-voltage switching, relying instead on star-point regulators and contactors in the primary lines; the elimination of the requirement of fast restarting after a fault; the elimination of the wide selection range for accelerator voltage, etc.

None of these options has yet been tried. They will require the interactive development of sources and power systems.

To increase the efficiency and reliability of the neutral beam power system, there is a need to provide direct recovery, determine the required regulation on the various power supplies, establish a realistic fault-energy limit, and develop a best conditioning procedure. Neutral beam power supplies can be constructed for voltages >150 kV if needed, but the use of the high-power triodes to regulate the voltage and to limit fault energy would require development.

Impurity influx from injectors

Species analysis of the ion beams extracted from an ORMAK source and a fractional-area TFTR source has indicated the presence of up to 2% oxygen. There is also evidence that neutral beam injectors have introduced oxygen into 2XIIB and copper into PLT, but the injected impurity levels were too small to degrade plasma performance. Before a specific development program in this area can be justified, more information on beam impurity levels is needed from test stands and tokamak heating experiments.

Magnetic shielding

The stray magnetic field at the ion source must be restricted to ~ 1 G. Because the field outside the TF coils may be hundreds of gauss, magnetic shielding must be provided for the ion source, the beam acceleration region, and the beam extraction region. The neutralizer region must be shielded to reduce the stray field there to ~ 10 G. The D III and TFTR neutral beam system designs include additional finger shielding between the neutralizer and the bending magnet to reduce the stray field in this region while allowing high pumping speeds. The ETF beam lines may also require magnetic shielding of the direct recovery subsystem.

Negative ion beams

Consideration of negative-ion-based neutral beams is motivated by the potential advantages of power efficiency at $E_b \geq 160$ keV and by the potential option of efficient beam production at $E_b > 200$ keV (see Fig. 5.4). This advantage will be useful if higher energies are eventually thought to be desirable for penetrating to the center of denser plasmas or if the required number of beam lines is reduced by going to higher energies at the same beam current. However, the development of D^- systems is expected to be insufficiently advanced by 1983 to allow a commitment to a D^- neutral beam system for the ETF. Nevertheless, there remains the possibility of replacing one or more D^+ systems with a D^- system at some later date during ETF operation.

5.6.4 Gaps

The U.S. neutral beam R&D program is not geared for needs beyond TFTR and MFTF. Although there is some low-level effort on the more advanced needs of an ETF system, the emphasis and pace do not support a 1990 ETF. More effort is required in the development of high-current, 100-s, 150-keV D^+ beams, the development of direct recovery, the study of neutron-irradiation effects, the development of power supplies and the relaxation of power supply needs, the improvement of reliability, the assessment and reduction of maintenance and conditioning requirements, the development of higher D^+ fraction ion beams, and the development of D^- beams. Upgrading present beam development test stands and/or creating new test stands is required to carry out this ambitious program.

5.6.5 Recommended Programs

The following items are listed in approximate order of urgency.

- (1) Ion beam. The development of reliable high-current D^+ beams at 150 keV is required, with tests at lower beam current and lower efficiencies at higher voltages.
- (2) Pulse length. The development of 10-100-s pulse length capability is required.

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Fig. 5.4. Comparison of overall neutral beam power efficiencies for various production methods of negative-ion-based beams and a positive-ion-based beam with 60% efficiency of direct recovery.

(3) Direct recovery. The development of the direct recovery of unneutralized beam energy with at least 50% efficiency is required.

(4) Monatomic fraction. At least 90% D^+ fraction in the ion beam should be reliably attained.

(5) Radiation hardening. The adequate lifetime of all injector components against the expected neutron bombardment must be ensured. Neutron-irradiation programs for sensitive materials such as seals and accelerator grids should be initiated. Beam line pumping systems (cryo-panels or getter pumps) must be able to operate in the anticipated neutron flux. If it is determined that the ion source must remain out of the line of sight of fusion neutrons, then a development program resulting in the demonstration of magnetic deflection of high-intensity ion beams is required.

(6) Reliability and maintenance. Intensified work on neutral beam reliability and design modifications to facilitate remote maintenance will be required. A detailed assessment is needed to determine the reliability and maintenance requirements. To meet the reliability requirement considerable effort will be required to determine the best materials and assembly techniques to minimize the need for time-consuming conditioning of the ion source.

(7) Power supplies. Significant impact might be made on injector costs and operational simplicity by investigating proposed schemes for the elimination of high-voltage regulation, high-voltage switching, and fast restarting after faults.

(8) Negative ion beams. If a certain amount of beam power at $E_b > 200$ keV is eventually found to be required to heat the center of denser plasmas, then high-current negative ion beams must be developed. At present no clear need for this capability is foreseen.

(9) Beam test stands. The most important task during the next few years will be the further development of beam injector parameters. An investigation should be initiated immediately to determine whether the 200-keV Beam Test Stand at LLL is adequate and available in time to pursue most of the above development items. By 1983-84 the problems of achieving adequate reliability and ease of remote maintenance will become the most vital issues.

5.7 AC POWER AND ENERGY STORAGE

5.7.1 Introduction

The ac power and energy storage for the ETF includes: (1) connections to existing utility lines, (2) ac distribution substations, (3) equipment providing large pulses of energy for startup, (4) special heating, (5) power for shutdown, and (6) the ability to dissipate or use generated electrical power produced by fusion.

The site selection and its facilities are important considerations in determining the needs for the ac power and energy storage requirements. For the purpose of this document it is assumed that the site will be adjacent to an ac power line whose deliverable output will be several hundred megavolt-amperes and whose voltage will be ~ 500 kV. Thus, the ETF will be connected to a relatively stiff system where ~ 1000 MW of pulsed power can be provided. It is also assumed that sufficient water will be available to dissipate several hundred megawatts of steady power and the energy dump of the superconducting systems in an emergency as well as during normal deenergization.

It would be desirable for the site to be capable of supporting a future power-generating system. This would then allow upgrading of the facility for future fusion development.

Although it may be possible to utilize the power utility grid to provide the pulsed energy, it is envisioned that large rotating motor-generators will provide the bulk of the pulsed energy and that the utility system will directly furnish all of the steady-state loads and a small amount of pulsed power.

5.7.2 Present Programs/Status

The present fusion ac power and energy storage facilities under construction are being completed with only minor RDAC's associated with ensuring that fatigue failure does not result from the large number of pulses. Fatigue on the ETF should not be a problem because of the long burn time. The number of full-load pulses will not exceed those on existing or presently planned facilities.

To utilize the utility power grids to the maximum and to minimize the on-site motor-generator energy storage system, the limitations on voltage fluctuations at a very low occurrence rate and on harmonics due to conversion rectification would have to be determined. Because of the difficulty of establishing such limitations, the trend has been to provide sufficient energy storage motor-generator flywheel (MGF) units to ensure that voltage fluctuations and harmonics do not become objectionable.

5.7.3 ETF Needs

The R&D needs for ac power and energy storage for the ETF are minimal. The utility and industrial progress in material and components is sufficient to meet the requirements for power and energy storage.

5.7.4 Gaps

Although engineering design has shown that the fatigue failure in the pulsed MGF energy storage systems is not difficult to prevent, the latest designs have not operated or do not have sufficient operating experience to prove out the designs. If the existing fusion program MGF units experience fatigue problems, special RDAC's may be required for the ETF.

5.7.5 Recommended Programs

As noted above there is no need for R&D or RDAC's for ac power or energy storage associated with the successful operation of the ETF. Because of the developmental nature of the ETF, there is a need for many nonrepetitive pulses of different rate, downtime, and maximum load. Therefore, it is not likely that the pulsed MGF energy storage system can be made as small as it might be for a steady power reactor of the same size.

The conceptual design engineering on the ETF should evaluate the capability of the utility system to provide as much pulse power as practical.

Because the pulses required by present devices are relatively short (lasting only a few seconds), they produce transients that are fast compared to normal utility loads. Mechanical shock would occur, and load-regulating equipment such as transformer tap changers would be cycled.

If, however, the ETF load could be increased gradually over a period of a few minutes and reduced in a similar manner, then the utility company could control their generator primary drive systems to provide the pulsed power. They would need to increase their spinning reserve to a minor extent, and they may desire a predetermined duty cycle; however, the need for a special MGF system could be minimized and possibly eliminated in the future. The fluctuation in the utility system grid could also be reduced by unloading variable-load equipment such as heating, air conditioning, and cooling systems during periods when the poloidal pulsed loads are high and also by providing maximum electrical power for these same loads during periods when the pulsed loads are off.

5.8 POWER CONVERSION

5.8.1 Introduction/Description

The power conversion system includes the rectifiers/converters, controls, switching, and energy dump circuits associated with the superconducting coils (TF and external poloidal coils) and the pulsed copper poloidal coils.

The present ETF base design provides for rf power to initiate the plasma and raise its temperature above the radiation loss plateau such that an OH circuit providing ~ 10 V/turn will be sufficient to heat the plasma to a value where neutral beam power can be utilized to reach ignition. By this process the need for the capacitors, OH switches, and switching resistors now utilized on present tokamak machine designs can be eliminated.

5.8.2 Present Programs/Status

The rectifiers/converters required for the superconducting circuits should be similar to those designed for use on existing devices. The currents and voltages as well as the protective systems should not be significantly different from those encountered in the Large Coil Program (LCP) at ORNL. However, the energy will be larger than on the LCP, and there will be a desire to provide a more reliable supply and protective system.

The rectifiers/converters needed for the copper internal poloidal coils will be similar to those now utilized on the recent tokamak toroidal systems. As presently planned there will be no need for a capacitor bank or switches to produce several hundred volts per turn on the OH system for startup. The maximum voltage required will be only ~ 10 V/turn, which can be obtained in a straightforward manner from nearly standard rectifier power supplies. Because the rise time is comparatively long (8 s), the ratio of power to stored energy is much larger than that of the MGF equipment presently being designed. The pulse rate will be less than that required for present machines.

5.8.3 Gaps

The major deficiency in the proposed electrical conversion system is the lack of data or proof that a low-voltage startup with a long rise time with reasonable rf power is feasible. However, there are no major gaps in the electrical circuits for the proposed system.

5.8.4 Recommended Programs

The immediate need is to obtain verification and develop a rf power system that will provide energy during startup on existing tokamaks and to show that there is no need for a poloidal system (OH circuit or blip circuit) to produce hundreds of volts per turn. Until the above verification is achieved, the effort to produce OH switches should continue.

The second major need is to determine how to prevent disruption or how to prevent extremely high voltages from being induced in the ETF

coils. Protective devices will be needed if it is not feasible to prevent the possibility of high voltages resulting from disruption, plasma movement, or external electrical circuits on the ETF.

Consideration must also be given to shielding the poloidal windings from induced voltages. This may best be accomplished by placing low-impedance circuits near the plasma that do not interfere or absorb large energy during startup but provide a path for large eddy currents when there is a disruption. These circuits could assist in protecting the vacuum-chamber walls during a disruption as well as providing shielding for external coils.

In present tokamaks any fault in the coil system causes fault-detecting devices to operate a crowbar, resulting in the dissipation in the coil circuit of energy stored in the field. In the event of a high-resistance-type fault, this energy will be essentially dumped in the fault, increasing the damage. With improved fault-sensing equipment and a reliable system, the energy stored in the field could be inverted out of the system, perhaps resulting in an improved fault protection system.

Unless there is a need for the quick removal of coil energy due to an abnormal condition such as a fault, there is not sufficient justification for providing thyristor bridges such as those used in TFTR. The stored energy in the coils at the end of the 8-s startup time is a very small percentage of the total energy required by the system. A combination of diodes and thyristors as used in JET or diodes and switches may be advantageous.

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6. MAGNETICS SYSTEMS R&D NEEDS ASSESSMENT

6.1 MAGNETICS SYSTEMS SUMMARY

ETF magnetics can be divided into the TF coil system; the induction coil system, which drives the plasma current; the control and shaping coil system, which controls the plasma exhaust of impurities and spent fuel; and the cryogenic system, which provides the liquid helium environment for the coils and the refrigeration for the coils and the neutral beam cryopanel.

In the ETF the TF and induction coils must be superconducting. The control and shaping coils will be superconducting if (1) they are outside the TF coils and (2) they would consume appreciable power if they were not superconducting. The control and shaping coils inside the TF coils will be resistive. Furthermore, if any of these coils are inboard of the blanket and shield modules, they must utilize inorganic insulation and 100% remote-handling techniques. The divertor coils will be resistive if an internal poloidal divertor is chosen. If a TF divertor such as a bundle divertor is chosen, tradeoff studies will be required to determine whether a resistive or superconducting version is preferable for the ETF.

6.1.1 Present Programs

The largest single program within the magnetics activity is the LCP at ORNL. Six d-shaped coils 2.5 m $\times$ 3.5 m in scale will be tested in a toroidal array at fields up to 8 T. Three coils are being built in U.S. industry, and Germany, Japan, and Switzerland are each supplying one coil. The coils will be subjected to various tests including pulse fields, simulated nuclear heating, out-of-plane loads, and emergency energy discharge. Five coils utilize NbTi, and one coil uses Nb₃Sn. Three coils use pool-boiling cooling, and three employ forced-flow cooling.

A second major program for the TF coil system is the 12-Tesla (12-T) Program, in which full-scale conductors are to be used to build

relatively small 0.4-m-ID by 1.0-m-OD test solenoids for testing in the High-Field Test Facility at LLL. The small size probably precludes using large-coil construction techniques but will be a necessary part of the testing of full-scale conductors. Four projects are included, one for NbTi at temperatures below 4.2K and three for Nb₃Sn. All of the U.S. manufacturers involved in the LCP program are also involved in the 12-T Program to ensure the scalability of the concepts used.

The energy storage and transfer program is directed primarily toward induction coil needs. The principal program, located at LASL, centers around the construction and testing of a 20-MJ prototype induction coil and the switching systems required for operation. There is parallel work at ANL under a much smaller program. There are also small programs at the University of Texas, the University of Wisconsin, and ANL on energy storage and transfer from superconducting coils or superconducting homopolar machines. The overall scale of the programs supporting the induction system is very much smaller than those supporting the TF coil system partly because the induction systems in some of the TNS studies were resistive; thus, the need for a superconducting technology for induction systems was not clear-cut until the ETF mission was defined.

There are various material evaluation programs, largely in direct support of the LCP or the MFTF. The National Bureau of Standards (NBS) is also carrying on relatively small evaluation programs in general support of the program.

There are no specific programs for the development of control and shaping coils, particularly demountable coils, although experience on present-day tokamak projects like PDX gives some indication of the problems to be faced.

There are also no specific programs to develop divertor coils and targets suitable for the ETF mission. There are poloidal divertor experiments under way (PDX) and bundle divertor experiments planned (ISX-B), but the coil technology used in these experiments is not directly applicable to the ETF.

6.1.2 ETF Needs

A number of specific ETF magnetic needs are listed in Tables 6.1-6.6 together with an assessment of the adequacy of the 1984 data base available for design decisions to be made and detailed design to follow. Some of the broad program areas from those tables are given in the time schedule charts (Figs. 6.1-6.5), one for each major magnetic subsystem.

The basic philosophy followed is that most key information must be available early in the design process, typically by 1985, requiring that considerable mockup and prototype construction take place prior to that date and before full-scale production begins. Although major programs like the LCP will contribute much to the coil concepts, the problems of the full-scale construction of coils and particularly cryogenic enclosures cannot be left until full-scale production begins. There is too much interaction with the entire design process. Ideas must be put to test in time for design alterations to be made.

6.1.3 Data Base Assessment and Program Gaps

The adequacy of the 1984 data base is assessed in Tables 6.1-6.6, and specific comments on each of the identified programs are made. In the tables the most applicable 1979 program responsibility is identified, the priorities of the needs for those areas considered either minimal or not adequate are given, and the program elements are identified as either design study needs or experimental needs.

The measure of the adequacy of the programs is whether they will provide an adequate base for firm design decisions to be made and allow detailed design to take place. The program needs are evaluated as to whether programs currently under way are likely to yield an ETF data base that is adequate (A), minimal (M), or not adequate (NA). The not-adequate category is intended to identify programs not receiving attention at a sufficient level to allow key decisions to be made on ETF magnetics. In those areas considered not adequate, programs must be initiated. The minimal category represents programs in which a higher level of activity would reduce the risk involving decisions based on these programs. The

adequate category represents programs that are receiving, or are scheduled to receive, attention at an adequate level.

6.2 TF COIL SYSTEM

6.2.1 General Comments

INTOR/ETF assessments place the peak field requirements for TF coils in the 10-12-T range, and for coil inner envelopes at approximately 6 by 10 m. Thus, the fields are higher than those for the LCP by 25-50% and the sizes larger by a factor of ~ 2.5 . Nevertheless, the LCP program is considered adequate to develop the basic data base for large superconducting coils. It has more recently been supplemented by the 12-T Program, where full-size conductors will be used to build 40-cm-bore inserts for the High-Field Test Facility at LLL. However, the Livermore facility can be considered only a minimal test because the insert coils are not subjected to realistic stresses, nor are large-coil techniques used in their construction.

6.2.2 Specific Comments

The following comments are keyed to Table 6.1, which lists a number of program elements necessary for the TF coil system. A schedule for these elements appears in Fig. 6.1.

(1.1) Niobium-titanium at 8 T will be covered in the LCP program. Construction techniques suitable for scale-up to the ETF are being utilized. The program is heavily weighted toward NbTi (five coils out of six) and operated at 4.2K or higher. It is thus not an adequate program for fields in the 10-12-T range.

(1.2) The use of NbTi at fields in the 10-12-T range requires the use of helium below 4.2K. The only program currently funded is the GA 12-T insert coil for LLL. It is considered that this system will provide only a minimal data base for the ETF. A larger test should be undertaken if this option is chosen. Consideration should be given to the conversion of one of the LCP coils to this mode in a later phase of that program.

Table 6.1. TF coil system R&D needs assessment

		1979 program	1984 data base ^a	Priority ^b	DS ^c
(1.1)	NbTi at 8 T (~4.2K)	LCP	A		
(1.2)	NbTi at 10-12 T (<4.2K)	12-T	M	M	
(1.3)	NbTi cooling options	LCP/12-T	A		
(1.4)	Nb ₃ Sn at 8 T	LCP	M	M	
(1.5)	Nb ₃ Sn at 12 T	12-T	NA	H	
(1.6)	Nb ₃ Sn cooling options	12-T	M	M	
(1.7)	Pulse-field loss measurements	LCP	A		
(1.8)	Pulse-field loss reduction		NA	M	
(1.9)	Protection from disruptions		NA	H	X
(1.10)	Structural properties (static)	LCP/MFTF	A		
(1.11)	Structural properties (fatigue, radiation)	NBS	M	M	
(1.12)	Neutron effects (insulators, stabilizers, sensors)	Misc.	A		
(1.13)	Modularization/remote RDAC's		NA	M	
(1.14)	Cryogenic enclosure EDAC's		NA	H	
(1.15)	Manufacturing scale-up RDAC's		NA	H	
(1.16)	Costing studies	ETF	M	M	X
(1.17)	Ripple control with trim coils	ETF	NA	H	X
(1.18)	Trim coil MFG RDAC's		NA	M	
(1.19)	Protection studies	LCP	M	M	X
(1.20)	Maximum credible fault requirement	BNL	M	M	X
(1.21)	Prototype ETF TF coil		NA	H	

^aA = adequate, M = minimal, NA = not adequate.

^bH = high, M = medium, L = low.

^cDS = design study as opposed to experimental R&D.

I

(1.3) The LCP project utilizes three coils with pool boiling and three coils with forced cooling. This is considered adequate, subject only to the limitation of (1.2) above.

(1.4-1.6) Only one of the LCP coils will utilize Nb₃Sn. If Nb₃Sn is chosen for the ETF, this single large-coil experience must be considered only a minimal data base. If peak fields in the ETF require Nb₃Sn at 12 T, the Livermore test, because of its small scale, cannot supply even a minimal data base. Some further scale-up exercise should be considered. One such opportunity exists in the Japan Cluster Test at the Japanese Atomic Energy Research Institute, which has 1 m x 2 m test coils and which will have a 12-T capacity.

(1.7-1.9) Pulse-field measurements of adequate scale are being performed in support of the LCP and will be performed in the LCP. However, insufficient attention is being given to reducing the effect of pulse fields on the TF system, particularly during disruptions. Consideration should be given to adding such a task to the induction system programs, where experimental studies of shielding potential coil regions outside the test induction coils could be undertaken.

(1.10-1.11) Structural experimental work to support the LCP, the MFTF, and the large MHD magnets is either under way or scheduled. The work on static properties can be expected to provide an adequate data base for the ETF and will, of course, be extended in support of the construction program. The TF coils will be subjected to a certain amount of cyclical loading as well because of the overturning-moment loading. The fatigue properties are generally less well covered and must be regarded as providing a more minimal data base.

(1.12) The level of radiation reaching the TF coils in the ETF mission is relatively low. The avoidance of insulators with known poor radiation properties may be sufficient. Some attention to the effects of various sensors on the calibration may be required.

(1.13-1.14) It will be important to translate ETF design concepts into dummy construction activities at an early stage. For example, the LCP has used a bell-jar approach with no individual dewars. Construction and testing of suitable enclosures with appropriate remote-handling details prior to final design is an absolute must. These represent new

programs not currently scheduled. It is possible that they can be explored on the LCP scale rather than full scale, possibly deriving some benefit from integration with LCP coils at some future time.

(1.15-1.16) Scale-up dummy construction will be a vital part of the ETF overall activity prior to the Title I Phase. Certain key elements of the TF coil, like structural cases, subplates, and cryogenic enclosure details, will profit from full-scale dummy construction activities. They will also add to the credibility of cost studies.

(1.17-1.18) Only conceptual attention has been given to the use of overcoiled TF coils and removable trim coils to reduce ripple. No real point design has been done on how such coils would be manufactured, supported, or removed. These designs need to be done, and presumably some dummy construction activities will be required if the concept is chosen and if critical areas are identified by those designs.

(1.19-1.20) Protection through the removal of magnetic energy to external dumps has received and will receive sufficient attention to qualify for a minimal data base for the ETF. Experimental studies are assumed to be carried out on the LCP. However, the LCP represents a rather simple environment relative to the ETF because there are few eddy current or breakdown paths. Decisions must be made on the maximum credible fault requirement; for example, must it be assumed that a full TF coil field will fail? If any less stringent requirement is used, then careful studies to justify such a requirement must be undertaken.

(1.21) The production of a prototype TF coil for the ETF should be started before production of the bulk of the coils begins. The main purpose would be to work out manufacturing, shipping, and field assembly techniques. Various test programs would be carried out but would not include a test of background field coils.

6.3 INDUCTION (OH) COIL SYSTEM

6.3.1 General Comments

INTOR/ETF assessments require a superconducting induction (OH) system. Field swings from +7 to -7 T at a rate of 14 T/s are a reasonable target for NbTi conductors and would satisfy the ETF requirements.

The development program in place is very much smaller than the TF coil program despite the fact that many of the problems faced by the induction system are more complex. There is inadequate program development and coverage to cope with unforeseen technical difficulties or alternate design opportunities.

6.3.2 Specific Comments

The following comments are keyed to Table 6.2, which lists the program elements required for the ETF induction coil system. The schedule for these elements appears in Fig. 6.2.

(2.1) One program to develop a 20-MJ, 50-kA coil is under way at LASL, and programs to develop some 10-MJ modules are under way at ANL. These programs must be considered secondary when judged against the TF program and the problems to be faced there.

(2.2) The ETF will require energies in the 500-MJ range, a large extrapolation from 20 MJ. Consideration must be given to some larger construction project, preferably one in which the modules are full scale and subjected to full stress.

(2.3-2.4) The largest conductor presently under development is 50 kA; this is felt to be the practical limit for pulsed cables. This appears to match ETF expectations provided that rf startup proves feasible. If faster initial rates of induction are required, it will be necessary to operate several 50-kA circuits in parallel. This problem should be addressed in the LASL 20-MJ program.

(2.5) Low-loss cables will require mixed-matrix stabilizers. Copper matrix cables are presently being used because of technical difficulties with the manufacture of mixed-matrix material. Increased development will be required to obtain the distinctly more optimum cables.

(2.6-2.7) The induction system must make a full field reversal on each pulse, which will place a high demand on the cyclical properties of the cables and support structure. The cables are, by nature, rather spongy, and the support structure must be nonmetallic. Neither of these programs is being addressed on an adequate scale to meet ETF milestones.

Table 6.2. Induction coil system R&D needs assessment

		1979 program	1984 data base	Priority	DS
(2.1)	Small-scale test (20 MJ, ± 7 T)	LASL/ANL	M	M	
(2.2)	Larger-scale tests (>100 MJ, ± 7 T)		NA	H	
(2.3)	50-kA conductor development	LASL/ANL	M	M	
(2.4)	Parallel circuit operation (>50 kA)	LASL	M	M	
(2.5)	Mixed-matrix cable development	LASL	NA	M	
(2.6)	Cyclical stress in cables	NBS	NA	H	
(2.7)	Cyclical stress in structures		NA	H	
(2.8)	Insulating cryogenic enclosures	LASL	NA	H	
(2.9)	Insulating support system tests		NA	H	
(2.10)	Manufacturing scale-up RDAC's		NA	H	
(2.11)	Switching systems	LASL	M	M	
(2.12)	Prototype ETF induction coil		NA	H

Fig. 6.2. ETF induction coil system.

Major programs on cyclical loading of cables and support structures and integrated magnet cable mockups will be necessary.

(2.8-2.10) The main induction coil requires that there be no electrically closed surfaces in the immediate vicinity, creating a problem for cryogenic enclosures and structural supports for both the induction and TF systems. This area has received very little design attention and essentially no experimental attention. Present induction programs will utilize simple cylindrical nonmetallic dewars, but these are far from adequate for the complex requirements of the ETF assembly. Major manufacturing and testing of large-scale enclosures and structures will be required. These should be at least LCP scale.

(2.11) The induction systems will require many parallel switch paths but will not require technology that is not being adequately explored on present-day nonsuperconducting induction systems.

(2.12) A prototype module of the main ETF induction system should be carried out before full production begins in order to work out manufacturing technique and to allow a qualification test program to be carried out.

6.4 CONTROL AND SHAPING COIL SYSTEM

6.4.1 General Comments

The INTOR/ETF assessments require that the coils necessary to maintain equilibrium and to properly shape the elongated plasma be located both inside and outside the TF coil. The outside coils, which would be superconducting, would present relatively few problems and are to be greatly preferred over inside coils in any tradeoff studies. The coils inside can be only copper coils. The internal coils, if outside the blanket and outboard of the plasma, can be relatively conventional but will require engineering development of remote handling and jointed assembly. If located inboard of the shield for control purposes, they will be required to operate in an extreme radiation environment. Coils inboard of the plasma are not likely to be required and should be avoided at all costs.

6.4.2 Specific Comments

The following comments are keyed to Table 6.3, which lists the program elements for the control and shaping coil system. The schedule for these elements appears in Fig. 6.3.

(3.1) The external coils are relatively simple but will require either nonmetallic or interrupted cryogenic enclosures.

(3.2-3.4) Coils inboard of the TF coil but outside the shield can utilize relatively conventional radiation-resistant insulators but will require considerable engineering development of the joints and remote-handling techniques. Full-scale mockups will be of particular use. Hydraulic loading frames can be used to simulate magnetic loads, a technique being widely used on JT-60 in Japan.

(3.5) Coils inboard of the shield, to be used only if no alternative solutions exist, must operate in an extreme environment. Presumably ceramic insulators would be suitable, but there is no experience with building and operating magnets with such insulators. Coils should be built and tested under magnetic loading and simulated heating.

(3.6) Internal coils with joints should be designed with the minimum number of turns. There is little experience with such high-current coils and with the current feeds. A program of construction, testing, and remote-handling techniques will be required.

(3.7) Preliminary studies have been done on the tradeoffs of internal vs external coils. Although external coils require more energy and greater demands on overturning-moment structure, they are to be preferred to internal coils when possible. More tradeoff studies should be carried out.

(3.8-3.10) The determination of the electrical coupling between the plasma, the metallic environment, and the control and shaping coils is a complex problem. Much analytical work can be done, but experimental measurements on realistic-scale mockups will be required.

(3.11) Plasma disruptions are likely to cause troublesome voltages in the control and shaping windings as well as major collapse pressures in metallic walls. Serious design studies should be carried out to see if this problem can be alleviated by additional coil systems, and

Table 6.3. Control and shaping coil

System R&D needs assessment		1979 program	1984 data base	Priority	DS
(3.1)	External superconducting coils	ETF	A		
(3.2)	Internal coil insulators		NA	H	
(3.3)	Internal coil joints		NA	H	
(3.4)	Internal coil remote assembly/ disassembly		NA	H	
(3.5)	Radiation effects on insulators (electrical field)		NA	M	
(3.6)	Segmented coils at high current		NA	H	
(3.7)	Inside/outside coil location studies	ETF	M	H	X
(3.8)	Coupling codes	ETF	M	M	
(3.9)	Electrical characteristics of environment	ETF	NA	M	
(3.10)	Time scale requirements	ETF	M	M	
(3.11)	Reduction of disruption voltages	ETF	NA	H	
(3.12)	Prototype internal coils		NA	H

Fig. 6.3. ETF control and shaping coil system.

tradeoff studies of any such potential solutions should also be conducted. If these coils are inside the shield, they must face an extremely difficult environment and will require considerable development.

(3.12) Prototype internal coils should be constructed well ahead of full production to allow reliability tests and to allow remote-handling techniques to be fully developed. Electrical and mechanical loading tests should also be performed.

6.5 DIVERTOR SYSTEM

6.5.1 General Comments

The INTOR/ETF assessment has determined that there will not be a sufficient data base for the ETF to ensure successful operation without a divertor. Compared to other magnetic systems divertor designs have received relatively little attention. If a reasonable design point can be found, a major new development program will be necessary to bring divertors anywhere near the stage of the reliability of the rest of the magnetics. The overriding consideration in a divertor choice must be the ability to provide remote maintenance. Although a bundle divertor is clearly more accessible, its effects on the plasma have not received the attention given to poloidal divertors.

6.5.2 Specific Comments

The following comments are keyed to Table 6.4, which lists the program elements for divertors from a magnetics point of view. The schedule for these elements appears in Fig. 6.4.

(4.1-4.2) Bundle divertor point designs must be carried out for the ETF. Trade studies would be done on copper coils, presumably requiring several hundred megawatts of power input, vs superconducting coils, with their greater shielding problem and complex environment. If superconducting coils are judged feasible, a major program of design, development, and testing will be required to ensure adequate reliability. The peak field and current density required will probably be comparable

Table 6.4. Divertor system R&D needs assessment

		1979 program	1984 data base	Priority	DS
(4.1)	Bundle divertor copper coils	ETF	NA	H	
(4.2)	Bundle divertor superconducting coils	ETF	NA	H	
(4.3)	Bundle divertor ripple requirements		NA	H	
(4.4)	Bundle divertor remote-handling RDAC's		NA	M	
(4.5)	Poloidal divertor remote designs		NA	H	X
(4.6)	Poloidal divertor remote-handling RDAC's		NA	M	
(4.7)	Targets		NA	H	
(4.8)	Target remote-handling RDAC's		NA	M	
(4.9)	Prototype divertor		NA	H

to that in the TF coil throat, but the out-of-plane structural requirements will be more severe. A divertor test facility should be seriously considered.

(4.3) The bundle divertor design is highly dependent on the allowed ripple in the undiverted area. Experiments on ISX-B should make this determination a priority.

(4.4) The bundle divertor, like many other major subsystems, must be removable by remote-handling techniques. Full-scale mockups will be a necessary part of the detailed design phase of the project.

(4.5-4.6) Poloidal divertor designs in current experiments are not suitable for remote-handling assembly/disassembly. Although segmented designs may be suitable, they have received little attention. Point designs, considerable engineering development, and mockup programs will be required to assess any designs deemed feasible.

(4.7-4.8) Pumping targets for divertors in current experiments do not have to cope with the high loads of an ETF target. Programs to test such loads, perhaps utilizing neutral beams for surface loading, will be required. Target studies would naturally be integrated with a divertor test facility.

(4.9) Prototype divertor elements should be constructed as early as possible to ensure the mechanical and electrical viability of the chosen concept. If the divertor is relatively easily replaceable as a unit, it can come relatively later in the program.

6.6 CRYOGENIC SYSTEMS

The following comments are keyed to Table 6.5, which lists the program elements for the cryogenic systems from the magnetics point of view. The schedule for these elements appears in Fig. 6.5.

(5.1) If forced-flow cooling is adopted for the ETF TF coils, attention should be given to the development of suitable pumps. The LCP coils are being driven by the refrigerator compressors, but this option is less attractive thermodynamically. Although considerable background exists, pumps of the required capacity and reliability are not commercially available.

Table 6.5. Cryogenic system R&D needs assessment

	1979 program	1984 data base	Priority	DS
(5.1) Forced-flow pumps		NA	M	
(5.2) Refrigerator controls during cooldown	LCP	NA	M	
(5.3) Refrigerator instrumentation	LCP	NA	M	
(5.4) Cryogenic enclosure RDAC's		NA	H	
(5.5) Reduced temperature systems ($<4.2K$)	12-T	M	M	
(5.6) Cryopanel systems	TFTR/MFTF	A		
(5.7) Prototype cryogenic enclosures		NA	H

Fig. 6.5. ETF cryogenic system.

(5.2-5.3) Commercial refrigerator systems are underinstrumented and have few automatic controls. Cooldown of massive loads with the attendant long time constants will require more diagnostics and controls than in the past. Such a development program could be integrated with a later phase of the LCP, with smaller-scale development work being carried out on existing refrigerator systems.

(5.4) The LCP has chosen a relatively simple cryogenic enclosure compared to the individual dewars required for the ETF. Past experience suggests that major dummy construction and test programs should be carried out in this area. If done on the LCP scale, they could perhaps be integrated into a later phase of the LCP.

(5.5) If NbTi at temperatures below 4.2K is selected for use in the TF coils, considerably more attention must be given to the refrigeration system for this range. There are undoubtedly more optimum systems than current-day refrigerators adopted for use at reduced temperatures.

(5.6) The cryopanel refrigeration needs for neutral beam pumping should be integrated with the coil refrigeration requirements.

(5.7) Prototype cryogenic enclosures should be constructed as early as possible in the program in order to develop concepts for assembly and a suitable quality assurance of vacuum and cryogenic isolation.

6.7 INTERFACES WITH THE MAGNETICS SYSTEMS

The R&D activities in areas that impact on the magnetics systems listed in Table 6.6 are as follows:

- (1) plasma systems,
- (2) electrical systems,
- (3) remote assembly,
- (4) blanket and first wall,
- (5) materials, and
- (6) safety and environment.

Table 6.6. Interfaces with magnetic systems

	1979 program	1984 data base	Priority	DS
Plasma systems				
Initiation and startup requirements	ETF	NA	M	
Programs to reduce requirements (rf initiation, etc.)		NA	H	
Shutdown requirements	ETF	M	M	
Disruption mitigation	ETF	NA	H	
Disruption time scale	ETF	NA	M	
Burn control requirements	ETF/ITR	M	H	
Degree of compression required	ETF/ITR	M	M	
Tolerable ripple	ETF	M	M	
Purposeful ripple (injection)	ISX-B	M	M	
Bundle divertor ripple allowance	ISX-B	M	H	
Inside/outside coil tradeoff	ETF	M	H	X
Electrical systems				
Protection of toroidal field	LCP	M	M	
Parallel circuit OH operation	LASL	M	M	
Intercoil system coupling	ETF	M	H	
Advanced concept energy storage (superconducting)	ANL	A	L	
Remote assembly				
Mockup RDAC's (buswork, etc.)		NA	H	
Dummy construction internal coils		NA	H	
Modularity studies	ETF	M	M	
Cryogenic enclosure mockups		NA	H	
Remote leak detection		NA	M	
Blanket/first wall				
Electrical characteristics	ETF	NA	M	
Access requirements	ETF	M	M	X
Leakage to TF coils (divertor)	ETF	M	H	X
Inboard shield requirements	ETF	M	M	X

Table 6.6 (continued)

	1979 program	1984 data base	Priority	DS
Materials				
Radiation properties of super-conducting coils	Misc.	A	L	
Suitable radiation-resistant insulators	Misc.	M	M	
Safety and environment				
Protection	LCP	M	M	
Code applicability	BNL	M	M	
Credible accident analysis	BNL	M	M	
Biomagnetic	DOE	M	M	
Access during hydrogen phase	ETF	M	M	

We have evaluated the data base likely to be available for the ETF in these areas as adequate (A), minimal (M), or not adequate (NA) but have not commented specifically about each element as we did in other sections.

7. SYSTEMS ENGINEERING R&D NEEDS ASSESSMENT

7.1 SYSTEMS ENGINEERING SUMMARY

Systems engineering R&D related concerns and their potential impacts on the ETF are summarized in Table 7.1. As shown in the table these concerns encompass three primary categories:

- (1) reliability, availability, and maintainability (including remote maintenance considerations),
- (2) environment and safety, and
- (3) materials data development.

Following is a brief discussion of these concerns and related recommendations addressing each of them. A more detailed treatment follows the summary.

7.1.1 Reliability, Availability, and Maintainability (RAM)

The concern regarding RAM involves the ETF Mission Statement requirement that the ETF device have an overall availability of 50% after the first two years of D-T operation. Achieving this goal in the most cost-effective way requires that realistic alternative apportionments of availability levels among the ~30 device equipments be carried out.

The two major elements that dictate device equipment availability are reliability and maintainability. The specification of realistic availability levels, therefore, requires a knowledge of the reliability and maintainability characteristics of the major subsystem components making up the overall device.

At present the data base regarding the reliability and maintainability characteristics of equipment peculiar to fusion devices (neutral beam injection systems, superconducting magnetic coils, torus vacuum vessels, etc.) is insufficient to obtain the needed characterization. To remedy this situation it is necessary to establish a data base derived from actual tokamak operating experience.

Table 7.1. Systems engineering R&D needs

Areas of need ^a	Impacts on ETF
Reliability/availability	Capital and operating costs, test-time availability
Remote maintenance	Design complexity and costs, availability, reactor relevance
Environment and safety	Costs of safety-related systems, ability of ETF to meet regulatory limits
Materials	Cost and life of hard-to-maintain elements

^aAreas receiving inadequate effort to support the ETF.

Accordingly, it is recommended that a program be established for the systematic collection of failure and corrective action data from ongoing tokamak programs. These data will serve as a basis for establishing factual reliability characteristics, cost-effective design approaches and levels of redundancy, and hands-on/remote maintenance requirements. Additional recommendations with respect to implementation of this program follow the summary.

7.1.2 Environment and Safety

It is recognized that the design of the ETF may require environment and safety R&D. It is expected that many of these R&D needs are similar to or the same as those discussed in other sections of this report or that they will be addressed by presently planned research programs such as TSTA, LCP, TFTR, and the fusion safety research program at Idaho National Engineering Laboratory (INEL). However, it is necessary to identify those ETF-specific needs not being met by these programs. A future R&D document will contain a detailed assessment of ETF environment and safety R&D needs.

7.1.3 Materials Data Base

To ensure the timely progress of the ETF, designers will need preliminary materials data for the Conceptual Design early in 1981 and firm design equations that describe materials properties in 1984. Areas in which materials data projections in support of either the Conceptual Design or Title I Design efforts appear to be inadequate have been identified.

In most cases pertinent materials R&D programs do exist. In some instances, however, extensions in scope and/or redirection in emphasis are needed to make these programs supportive of ETF requirements. Anticipated ETF needs, together with an indication of their relative priorities, are summarized in the following paragraphs. The priorities noted refer to additional data needs; "low priority" means that existing programs are considered adequate, not that the programs themselves have low priority for the ETF.

Irradiation effects on ceramics

Two key applications in the ETF of ceramics having high-radiation exposure are circuit breaks in the first wall and armor/coatings for the first-wall superstructure. It is not clear whether current programs will provide adequate data for the selection of materials and component design. This is a high-priority effort.

High-conductivity radiation-resistant alloys or coatings

Broadcasting rf power into a plasma requires an antenna and/or a waveguide termination on the first wall that has good electrical conductivity and can withstand intense radiation. The current R&D program does not adequately address the combined requirements for good conductivity and radiation resistance. This is a high-priority effort.

Surface radiation effects on heat-sink materials

Current efforts on plasma/surface interaction are directed at plasma contamination. Also needed for the ETF design is information on the combined effects of ion bombardment and radiation on the structural integrity of first-wall components, especially heat-sink materials. This is a medium-priority effort.

Repair welds on irradiated components

The limitations of making repair welds on materials exposed to low and moderate levels of irradiation should be established and alternate methods developed as necessary. This is a medium-priority effort.

Compilation of data on martensitic steels

Pertinent data are presently spread among vendors, the European reactor industry, and U.S. fast breeder research. Additional data are forthcoming from a number of sources via the Fusion Reactor Materials

Program (FRMP). These data must be gathered and compiled to provide a useful data base for ETF design. This is a high-priority effort.

Engineering limits for divertors

There are materials limits (such as maximum heat deposition) that will impact divertor designs. The limitations of current divertor technology should be defined, and the potential for expanding these limits should be evaluated, including empirical tests on selected materials. This task has high priority.

Combined strain/radiation damage

Strain damage and radiation damage will combine to increase the resistivity of copper stabilizers in superconducting magnets. The system should be modeled using existing data on the annealing behavior of mildly cold worked copper. The testing of strained and irradiated copper may be necessary. This task has medium priority.

Streaming-path neutronics

The shielding calculations used during the Conceptual Design Phase to size components will probably not be adequate for defining radiation damage to components through streaming paths. Calculations on two to three streaming geometries should be performed to determine the expected range of radiation dose enhancement. These data would then be used to estimate radiation damage and component lifetimes. This task has medium priority.

Radiation damage in diagnostic systems

Probable placement of diagnostic hardware and preliminary materials selections should be defined to permit identification and/or avoidance of potential radiation damage problems. This task has low priority (with respect to the need for additional data from R&D programs).

The foregoing paragraphs have highlighted ETF needs in terms of materials-oriented R&D programs and design studies beyond the scope of the present ETF program. Detailed information on these subjects follows.

7.2 RELIABILITY/AVAILABILITY/MAINTENANCE

The ETF project requires a major shift in the emphasis of the supporting technology and confinement programs. Until now the emphasis of the fusion program has been on the need to develop scientific proof of principle and establish engineering feasibility. However, the series of new devices starting with the ETF must not only meet the design goals for physics performance but must also be capable of continued, long-term operation. This will add a new dimension to the programs because of the need to develop maintenance systems and components with high reliability. Therefore, to establish the technology base for the ETF, the emphasis of the program must shift from science to engineering.

An overall availability of at least 50% will be necessary for the ETF to provide the integrated test conditions for EPR and DEMO technology development. If this level of availability is to be achieved, a major emphasis on reliability and maintenance will be required in the design and development of ETF systems. For example, the ~30 major systems of the ETF facility will require an average availability of 98% for the total facility to reach 50% availability. On current machines many of the individual tokamak systems, such as the neutral beam injectors, probably have considerably less than 50% availability. However, there has not been sufficient collection and documentation of operations data to assess the state of the art. To begin the transition toward an emphasis on engineering, future programs will emphasize reliability as a design requirement, and the collection and documentation of data on failures and maintenance will be required in the test program.

Availability is a measure of the ratio between operating time and operating time plus downtime. As such it is a very important factor in determining system effectiveness and cost. The actual availability of the ETF will be a function of design and support. The design parameters

that determine availability are system reliability and maintainability. Reliability in design influences the frequency of system failures whereas maintainability in design influences the active repair time associated with each system failure. The actual repair time per system failure may be significantly greater than the active repair time. The difference can be caused by a shortage of spares, insufficient manpower, unavailability of support equipment, inadequate technical support or training, etc. All of these nondesign-related reasons are considered in the level of support. Clearly availability has to be a prime consideration in the design and support of a device if the actual availability in operation is to be optimized.

At present there is a paucity of data regarding the reliability and maintainability characteristics of equipment peculiar to fusion devices. Typical of this equipment are neutral beam sources, superconducting magnetic coils, large copper magnetic coils, torus vacuum vessels, tritium system components, getters, and limiters. In order to identify potential availability drivers and assign meaningful availability apportionments, it is necessary to establish a data base that is derived from the operating experience of actual tokamaks.

Accordingly, it is therefore recommended that a program be established for the systematic collection of failure and corrective action data from ongoing tokamak programs. The failure/corrective action program should encompass the establishment of a common format for data collection, an identified codification of equipment for purposes of filing and data retrieval, a computerized data storage system, and an identification of responsibility for collecting and reporting the data to the ETF project and throughout the fusion community.

The data required for assessing reliability characteristics would include

- (1) the equipment identification (via the standard equipment-coding system);
- (2) the number of operating hours or cycles on the equipment at the time of failure;
- (3) the phase in which failure occurred (i.e., manufacture, test, or operation phases);

- (4) the environment in which the equipment was operating (radiation level, magnetic field, temperature, and pressure);
- (5) the nature of the failure and the suspected cause; and
- (6) the location of the equipment within the machine or test cell.

The information required for evaluating the maintainability characteristics of the equipment would include

- (1) the time required to isolate the failure;
- (2) the time required to gain access to the failed equipment;
- (3) the time required to effect the repair or removal and replacement of the failed equipment, including the time it takes to obtain replacement parts;
- (4) the time required to verify system integrity after repair or replacement;
- (5) the manhours required to effect the repair or replacement;
- (6) special tools required to perform the repair or replacement; and
- (7) the cost of material and labor associated with effecting the repair or replacement.

The compilation of the foregoing information will result in a statistical data base on which to establish design approaches, cost-effective levels of redundancy, more accurate predictions of reliability and availability, and realistic maintainability criteria.

7.3 MATERIALS DATA BASE

7.3.1 Introduction

The development of materials data, especially the effects of neutron irradiation, is vital for the timely progress of the ETF. Designers will need preliminary data for the Conceptual Design Phase early in 1981, and beginning in 1984 firm design equations describing materials properties will be required.

A general report on the existing materials data base has already been compiled for INTOR, and that information will not be repeated here. This report deals primarily with specific materials applications in the

ETF where data projections for either the Conceptual (1981) or Title I (1984) Designs are less than adequate. In most cases appropriate materials programs exist, but extensions of scope or redirections in emphasis are needed. Existing programs are described briefly in Sect. 7.3.2.

Materials applications of concern in each major reactor system are presented in Sect. 7.3.3. Also given is a brief summary of high-priority needs for DOE programmatic support and for project design studies.

7.3.2 Present Programs

The major materials program supporting fusion is DOE's FRMP. Its mission is to develop materials for a first wall and applications such as insulators and heat-sink materials that must survive intense radiation.

Within the FRMP DOE recently initiated efforts to develop design applications and feasibility for using martensitic (ferritic) steels in the ETF. Fabrication processes, radiation effects, and fatigue and fracture behavior are the principal subjects of investigation.

An active liaison between the ETF project and the fusion materials community will provide coordination of ETF design needs and goals in the FRMP. (The recently initiated program on martensitic steels is a promising start.)

Several programs directed at component development for fusion reactors will also generate materials data. In the magnetics system, for example, the LCP involves the fabrication of coils using Nb_3Sn and NbTi superconductors. Also, NBS manages a DOE program on low-temperature materials applications for fusion that includes resolving the potential problem of weld failures in cryogenic magnet case materials.

The brief remarks above deal with the scope of materials programs. However, it is also important to recognize the difficulty of the programmatic goals, especially with regard to irradiation effects on components exposed directly to the plasma, i.e., the first wall and attached structures.

The ETF itself will provide the first realistic fusion environment for materials testing. Materials development supporting the ETF must be done in nonrepresentative sources, primarily in fission reactors.

Significant differences in potential materials damage from high-energy neutrons (fusion) and from fast neutrons (fission) are already known to exist. Some high-energy radiation effects in some materials can be simulated in fission reactors (for example, helium generation by the transmutation of nickel in stainless steels). However, it is likely that some predictions of irradiated materials properties will have low confidence. Certainly this will be true in the Conceptual Design Phase.

In parallel with the detailed design of the ETF (1985-1986), materials testing in FMIT will provide limited high-energy neutron data for the corroboration and recalibration of materials behavior projected from fission data. Thus, data and design equations will probably be revised several times during the Title II Design Phase, and confirmation of some design values may even extend into the construction phase.

The link between FMIT and the ETF is important because results from FMIT will confirm to the best degree possible design assumptions about materials performance in the ETF. Confidence in these design assumptions will affect the conservatism necessary in design, thus impacting cost, and will also affect evaluations of reliability. The support available from FMIT for the ETF design would diminish with any delay or decrease in the availability of FMIT or with an acceleration of the ETF schedule.

7.3.3 ETF Needs

This section highlights areas of materials technology where help from outside the ETF project will be necessary and where projected resources appear inadequate. Table 7.2 lists areas of research where current trends indicate that national programs will not supply adequate data. Table 7.3 lists areas where the projected data base is probably adequate but where significant effort using resources outside the ETF project will be necessary to apply the existing data to the ETF. The numbers in the "Reference" columns in Tables 7.2 and 7.3 refer to

Table 7.2. Program needs

Item	Priority	Reference
(A) Irradiation effects on ceramics	High	(1.5, 1.6, 1.9, 4.1)
(B) High-conductivity radiation-resistant alloys or coatings	High	(4.3)
(C) Surface/radiation effects on heat-sink materials	Medium	(1.5, 1.8, 2.1, 2.2, 6.6)
(D) Repair welds on irradiated components	Medium	(1.7)

Table 7.3. Recommended design studies needs

Item	Priority	Reference
(A) Compilation of data on martensitic steels	High	(1.2, 1.4, 3.1)
(B) Engineering limits for divertors	High	(2.1, 2.2, 2.3, 6.6)
(C) Combined strain/radiation damage	Medium	(3.4, 3.5, 6.2)
(D) Streaming-path neutronics	Medium	(3.5, 3.6, 4.1, 4.2, 6.2)
(E) Radiation damage in diagnostic systems	Low	(6.5)

items in Table 7.4 in the next section, where specific materials applications are discussed.

With respect to program needs (Table 7.2), items (A) and (C) lie within the present scope of the FRMP, and items (B) and (D) are not adequately addressed.

Item (A). Ceramics have second priority behind the development of first-wall alloys; consequently, resources for irradiation experiments on ceramics are quite limited. Two key applications of ceramics in the ETF involving high-radiation exposure are (1) circuit breaks in the first wall and (2) armor and coatings for the first-wall superstructure. It is not clear that the current programs will provide adequate data for the selection of materials and for component design. Even for poloidal coils where radiation levels are less and closer to a fission spectrum more data are needed.

Item (B). Broadcasting rf power into a plasma requires some type of superstructure on the first wall (either an antenna or a waveguide termination) that has good electrical conductivity and can withstand intense radiation. RF losses in the antenna/waveguide will add to its heat load from radiation. Simultaneous requirements for good conductivity and radiation resistance are not well addressed in current programs.

Item (C). Current efforts on plasma/surface interaction are directed at plasma contamination. Clearly the plasma physics questions deserve first priority; however, some measure of the combined effects of ion bombardment and radiation on the structural integrity of first-wall components, especially heat-sink materials [item (C)], will be needed for designing the ETF.

Item (D). The repair of materials exposed to low and moderate levels of irradiations in the ETF seems inevitable. Welding metals containing transmutation products such as helium are problematic. The limitations of repair welding should be established and alternate methods developed as necessary.

With respect to project design studies (Table 7.3), items (A) and (E) require research and organization of existing or forthcoming

data from many sources, and items (B), (C), and (D) are miniprojects requiring laboratory work.

Item (A). Pertinent data on martensitic steels are spread among vendors, the European reactor industry, and U.S. fast breeder research. More data are forthcoming from a variety of sources under the auspices of the FRMP. These data must be gathered and compiled for designers of the ETF.

Item (B). Divertor technology is in a state of flux, as yet lacking coherence. Apart from the physics questions there are materials limits such as maximum heat deposition that will impact divertor designs. The limitations of current technology should be defined, and the potential for expanding these limits should be evaluated, including empirical tests on select materials using lasers or electron beams.

Item (C). Strain damage and radiation damage will combine to increase the resistivity of copper stabilizers in superconducting magnets. The system should be modeled using existing data on the annealing behavior of mildly cold worked copper. Empirical tests on strained and irradiated copper may be necessary.

Item (D). The general shielding calculations used to size components during the Conceptual Design Phase probably will not be adequate for defining the radiation damage to components through streaming paths (for example, local damage to poloidal coil materials from streaming through shielding joints or damage to injector components from streaming through the injector orifice). Calculations on two or perhaps three streaming geometries should be performed to determine the expected range of the enhancement of the radiation dose. These models would then be used to estimate radiation damage and component lifetimes.

Item (E). The probable placement of diagnostic hardware and preliminary materials selections should be defined so that potential radiation damage problems can be identified.

7.3.4 ETF Needs/Details

The following comments are keyed to Table 7.4, which identifies the ETF needs in more detail. Table 7.4 does not include all major materials

Table 7.4. Materials applications of concern.

	Program ^a	1981 ^b	1984 ^b	Priority ^c	Design study ^d
First wall/structure					
(1.1) Chamber wall - 316 SS	FRMP	M	A	Low	
(1.2) Chamber wall - ferritic	FRMP	NA	A	High	X
(1.3) Joints/seals ^e	FRMP	NA	M	Low	
(1.4) Shielding - 316/ferritic	FRMP	M	A	Low	
(1.5) Limiter/armor ^e	FRMP	NA	M	High	
(1.6) Circuit break ^e	FRMP	NA	M	High	
(1.7) Repair welds		NA	NA	Medium	
(1.8) Surface damage	FRMP	NA	M	Medium	
(1.9) Coatings	FRMP	NA	NA	Medium	
Divertor					
(2.1) Bundle divertor collector ^e		NA	M	High	X
(2.2) Poloidal divertor collector ^e		NA	M	High	X
(2.3) Bundle divertor - copper		A	A	Low	X
Magnet systems					
(3.1) TF case ^e	LTM ^f	NA	A	Low	
(3.2) Superconductor NbTi/Nb ₃ Sn	LTM	M	A	Low	
(3.3) Organic insulators	FRMP	M	A	Low	
(3.4) Copper stabilizer		A	A	Medium	X
(3.5) Poloidal coils - copper		A	A	Low	
(3.6) Poloidal coils - insulator	FRMP	M	A	High	
Neutral beam/rf					
(4.1) Neutral beam ceramic insulator	FRMP	M	A	Low	
(4.2) Neutral beam grid/source		M	A	Low	
(4.3) RF antenna/waveguide		NA	A	High	
Blanket/test modules					
(5.1) Structural material ^e	FRMP	NA	M	Low	
(5.2) Breeder/coolant	FRMP	NA	M	Low	

Table 7.4 (continued)

	Program ^a	1981 ^b	1984 ^b	Priority ^c	Design study ^d
Other					
(6.1) In-vacuum requirements		M	M	Low	X
(6.2) Neutronics data base/streaming	NDEF	M	M	Medium	X
(6.3) Advanced materials testing requirements		M	A	Low	X
(6.4) Materials quality assurance — leak detection		NA	A	Low	
(6.5) Diagnostic system materials			A	Low	X
(6.6) Heat deposition limitations		M	M	Medium	X
(6.7) Cryopump panel coatings		NA	A	Low	

^aMaterials programs with current or projected effort in the specified application.

^bThe adequacy of the projected data base for the Conceptual Design (1981) and the Title I Design; A = adequate, M = minimal, and NA = not adequate.

^cThe relative priority of need for additional data as noted in comments; priority reflects both the immediacy of the need and the importance of the data to the overall design.

^d"X" indicates areas for project-directed studies involving resources beyond the ETF group.

^eMaterial to be determined.

^fLow-Temperature Materials Application Program (NBS).

applications in ETF; it includes only those applications where the adequacy of data to support the Conceptual and Title I Designs for the ETF is in question. "Adequate" here means that generally data will exist over the appropriate range of materials properties; a statistical data base, as might be needed to qualify a material for commercial service, is not implied. In many cases the projected data for detailed design are adequate, but the more immediate data needs for the Conceptual Design are lacking. Also, for some components it is assumed that solutions to materials problems will accompany future hardware development.

Projected materials data to support the detailed design of the ETF appear to be generally adequate based on optimistic assumptions about materials development in national programs and about hardware development. Priorities in Table 7.4 apply to additional data needs; low priority generally means that existing programs are adequate, not that these programs themselves have low priority for the ETF.

First wall/shielding

The first wall and shielding will be operated at relatively low temperatures, below the range where intense irradiation in fast reactors induces swelling in commercial ferrous alloys. Two types of materials are being considered for major structural components in the first wall and shielding: type 316 stainless steel (SS) and ferritic steels. The goal lifetime for first-wall components is 6 MWyr/m². Following are specific comments keyed to Table 7.4.

(1.1) A large data base exists on the properties of irradiated 316 SS and 20% cold-worked 316 SS. Needed additional data on fatigue and crack growth are currently being developed in the FRMP.

(1.2) The current data base on irradiated and unirradiated properties is weak, especially in the areas of fatigue and fracture. Recently the FRMP initiated efforts to develop a data base on ferritic alloys for the ETF. Welding techniques and weld properties will be an integral part of the development program.

(1.3) The requirements for joints and seals in the first-wall system are ill defined at this time. If joints and seals must be near the plasma, requirements for creep resistance, fatigue resistance, and high yield strength at a high temperature and under intense irradiation may well be more severe at the joints, where adequate cooling may be a problem, than at the first-wall surface.

(1.4) Protecting portions of the first wall or protrusions such as waveguides may require heat-sink materials exposed to intense radiation. The special purpose materials task in the FRMP has work in progress on the radiation effects in several heat-sink materials; however, the current data base is inadequate for materials selection and design.

(1.5) Radiation-resistant ceramic insulators are also being studied in the special purpose materials task of the FRMP. Although several promising candidates have been identified, current data are inadequate for materials selection and design.

(1.6) Repair welds on irradiated material present a materials problem as well as a technological problem. Welding causes the redistribution of transmutation products that could result in severely degraded mechanical properties of repair welds, e.g., helium at grain boundaries. No current programs address this problem.

(1.7) The surface degradation of armor, divertor collector plates, and first-wall alloys and its potentially damaging effect on structural integrity lies within the scope of the FRMP. However, the current emphasis in plasma/surface interaction studies is on plasma contamination. Thus, little data will be available in the near future on the structural impact of surface degradation.

(1.8) Coated-limiter studies in ISX and PLT will provide some data on coatings. However, radiation damage in coatings is not being investigated.

Divertor

Engineering requirements for a divertor in the ETF must await better understanding of the behavior of diverted plasmas. Generic materials problems expected in the divertor include thermal shock,

thermal fatigue, and surface damage from intense particle bombardment. Currently planned surface studies in the plasma/materials interaction task of the FRMP focus on the contribution of materials to plasma impurities. (Until the basic physics is defined, the engineering questions are secondary.) Thus, the future sources of materials data on surface damage that might impact the design of a divertor for a long-pulse, high-duty machine are not clear. A second potential problem is radiation damage in the divertor coils because of potential limitations on shielding. Following are specific comments keyed to Table 7.4.

(2.1) The development of divertor technology with somewhat relaxed materials requirements is assumed.

(2.2) The development of divertor technology with somewhat relaxed materials requirements is assumed.

(2.3) The prediction of resistivity change in copper coils due to irradiation damage will be limited in accuracy by uncertainties in neutronics data. Although the scope of the data base is adequate, reductions in uncertainty are desirable. The spectral sensitivity of damage and poor source characterization will probably not be resolved in the near future, resulting in conservative limits for shielding.

Magnetic systems

Low-level radiation damage to conductors and insulators limits the life of magnet components and is a direct result of the amount of shielding present. The real issue is the cost-effectiveness of increasing the radiation resistance of the magnet components. A marginal data base on irradiation effects exists now, and there is work in progress in the program.

Fabrication experience on large superconducting magnets is forthcoming in the LCP, and a materials data base is being compiled in the Low-Temperature Materials Application Program managed by the NBS. Following are detailed comments keyed to Table 7.4.

(3.1) Unacceptable fracture behavior of potential TF coil case materials at cryogenic temperatures is a serious problem under study in

the Low-Temperature Materials Application Program. A future solution combining improved welds and better design is assumed.

(3.2) The projected data base is considered adequate.

(3.3) The projected data base is considered adequate.

(3.4) The acceptable time periods defined between intermittent anneals of superconducting magnets will determine the amount of copper stabilizer needed and hence the size of the magnets. The combined effects of strain damage and radiation damage on residual resistivity after annealing copper are important factors. Although substantial information on radiation damage and annealing kinetics in copper exists, the combination of strain damage, radiation damage, and annealing procedures may require further tests.

(3.5) Although the data base is adequate in scope, reductions in uncertainties about radiation damage could reduce the amount of shielding needed [see (2.3) and (6.2)].

(3.6) The projected data base on insulators is considered adequate.

Neutral beam/rf heating

Neutral beam heating and rf heating do not inherently present materials problems. However, each has components with potential materials limitations. Following are specific comments keyed to Table 7.4.

(4.1) Vacuum-tight seals between the metal housing and the large ceramic insulators in a neutral beam injector require a ceramic-to-metal bond that has thus far proven problematic in fabricating the TFTR injectors. The resolution of this problem is assumed.

(4.2) Radiation damage to the injector grid and source has been identified as a potential life-limiting problem for these components. Better identification of materials requirements and available data is needed.

(4.3) If rf power is transmitted to the plasma using a waveguide, then the protruding waveguide termination will be exposed to a first-wall irradiation environment plus additional heating from rf losses.

Blanket/test modules

Blanket modules and materials test modules will require operating temperatures significantly higher than the temperature of the structural first wall of the ETF. Temperatures from 300-600°C are probable and will increase the radiation damage occurring in structural materials in the modules. However, the service life of the modules will probably be 0.5-2 MWyr/m² maximum, and they will be periodically removed for examination.

Specific comments keyed to Table 7.4 follow.

(5.1) Both now and in 1984 there will be insufficient data for selecting materials and designing test modules. However, by the late 1980's the development of advanced materials using results from FMIT will presumably provide an adequate data base.

(5.2) Adequate data are assumed to be forthcoming by the onset of the design of blanket modules.

Other materials concerns

The following comments are keyed to Table 7.4.

(6.1) If the primary vacuum boundary of the ETF extends beyond the basic torus and auxiliary equipment must operate in a vacuum, then a substantial program to identify special equipment and lubricants, working fluids, etc., suitable for use in vacuum coils will be necessary.

(6.2) Two basic problems in neutronics calculations are uncertainties in cross-section data and difficulty in modeling complex shapes. Both problems make defining local maximum damage rates from streaming problematic. Yet estimates of maximum damage rates will determine the shielding requirements and service lifetimes of many components.

(6.3) The amount of success of preliminary materials development for fusion in the mid-1980's may well determine the sequence of advanced fusion reactors necessary to proceed to a DEMO. Better definition is needed of the potential range in materials testing that will be required in the ETF. This information will be needed for project support and for preliminary designs of test modules.

(6.4) Leak detection methods for the ETF and especially for the first wall are not well defined but may impact materials quality assurance methods. The future resolution of leak detection methods and suitable quality assurance procedures are assumed.

(6.5) Diagnostic requirements for the ETF are not yet well defined. The increased exposure of diagnostic equipment, compared with low-duty experimental tokamaks, may cause radiation damage in some components.

(6.6) Maximum tolerable heat deposition rates may determine engineering limits in several heat-sink applications, for example, divertor collectors. Relevant data probably exist but need to be consolidated and applied to fusion reactor components. Some empirical tests may also be needed.

(6.7) Special coatings on cryopanel are necessary to trap helium. The recharging cycles of cryopumps present a potential fatigue problem for the coating. The resolution of this problem during the development of the cryopanel is assumed.

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