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FED-R

A FUSION ENGINEERING DEVICE UTILIZING RESISTIVE MAGNETS

FEDC STAFF
D.L. JASSBY and S.S. KALSI, Editors

FUSION ENGINEERING DESIGN CENTER

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Printed in the United States of America Available from
National Technical Information Service
U S Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
NTIS price codes—Printed Copy A14 Microfiche A61

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ORNL/FEDC--82/1

DE83 009765

ORNL/FEDC-82/1
Dist. Category UC-20

Contract No. W-7405-eng-26

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RESISTIVE MAGNETS

FEDC Staff
D. L. Jassby and S. S. Kalsi, Editors

Date Published - April 1983

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PREFACE

This document is one of four describing studies performed in FY 1982 within the context of the Fusion Engineering Device Program for the Office of Fusion Energy, Department of Energy. The documents are:

1. FED Baseline Engineering Studies
2. Considerations of an Advanced Performance Fusion Engineering Device - FED-A
3. FED-R - A Fusion Engineering Device Utilizing Resistive Magnets
4. Technology Demonstration Facility - TDF

These studies were designed to extend the studies on the FED Baseline and to develop innovative and alternative concepts for the FED.

This report describes the studies on FED-R, a device that utilizes resistive toroidal-field coils. A primary mission is to provide nuclear testing capability for a variety of blanket assemblies. The reference FED-R concept was developed and analyzed by the Fusion Engineering Design Center with D. L. Jassby of PPPL serving as Design Manager and S. S. Kalsi of the FEDC serving as Project Manager. This report presents the objectives and design requirements for FED-R and summarizes the design and analysis carried out in FY 1982.

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ACKNOWLEDGMENTS

We wish to acknowledge the dedication, skill and unflagging cooperation of Shirlene Dale, who provided the secretarial services essential to preparation of this report. We also wish to thank Paul Fogarty, Bill Kunselman, and Al Williams for the high level of cooperation in the preparation of many of the drawings and figures and Jane Parrott and staff for preparation of all the final artwork for the document. We express our thanks to the FED Reports Office for their assistance in final layout, proofing, and processing the document. Without their efforts, this report would not have been possible.

In the technical area, we would like to acknowledge the assistance of O. Anderson, W. Cooper, and K. W. Ehlers of LLNL, D. Mikkelsen of PPPL, W. T. Reiersen of FEDC/GAC, and J. Schultz and R. J. Thome of MIT. The authors appreciate the encouragement of T. Shannon, FEDC Manager, and C. A. Flanagan, FEDC Projects Manager.

This work was supported by the U.S. Department of Energy, Office of Fusion Energy.

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FED-R
A FUSION ENGINEERING DEVICE UTILIZING RESISTIVE MAGNETS

ABSTRACT

The principal purpose of the FED-R tokamak facility is to provide a substantial quasi-steady flux of fusion neutrons irradiating a large test area in order to carry out thermal, neutronic, and radiation effects testing of experimental blanket assemblies having a variety of configurations, compositions, and purposes. The design of the FED-R device also suggests potential for an upgrade that could be employed as a full-scale demonstration reactor for some specific fusion-neutron application when required.

The emphasis on reliable nuclear-testing capability demands that the plasma physics characteristics and technological features of the fusion machine be chosen as close to mid-1980s' state-of-the-art as possible. Hence the specified plasma parameters for initial (Stage I) operation are selected to be near those expected in TFTR or DIII-Upgrade, with the important exception that FED-R demands high-duty-factor operation using a single-null poloidal divertor for particle control and heat removal. Neutral-beam injection is used for plasma heating and current drive in the steady state, as well as for partial fueling and for enhancing fusion-neutron production.

The plasma has major and minor radii of 3.5 m and 0.85 m, respectively, with a vertical elongation of 1.5. The nuclear-test region is located on the outboard side of the plasma and is 80-cm deep with approximately 60 m² of exposure area. For flexibility, the test region is located outside the vacuum boundary. The outboard vessel wall is a stainless steel double wall with 20-atmosphere helium coolant that gives more than 80% transmission of 14-MeV neutrons with adequate heat removal and acceptable stresses caused by the atmospheric pressure load and a steady thermal load up to 60 W/cm².

The toroidal-field coils are fabricated of water-cooled copper plates and are sufficiently massive so that they can be operated steady

state with a power dissipation of 180 MW in State I (0.4 MW/m^2 at $Q_p = 1.5$) or 280 MW in a Stage II upgrade (1.3 MW/m^2 at $Q_p = 2.5$). The TF coils have demountable joints to facilitate access to all the in-bore components for maintenance and to allow installation of outsized nuclear assemblies. The demountable joints also permit an option for the Stage II upgrade wherein the coil bore is enlarged for installation of a new vacuum vessel and permit nearly complete blanket coverage for full-scale demonstration of a fusion-neutron application whose feasibility will have been verified in Stage I.

The total electrical power consumption of the FED-R facility is approximately 425 MW in Stage I and 600 MW in Stage II. The total direct cost for Stage I is estimated to be \$510 million, with an additional \$220 million of direct cost required for the Stage II upgrade.

1. MISSION AND OBJECTIVES

1.1 MISSION STATEMENT

The mission statement for FED-R is the following:

"The primary mission of FED-R is to provide a reliable quasi-steady 14-MeV neutron source of the order of 100 MW intensity and irradiating at least several tens of square meters of blanket module assemblies for the purpose of developing fast-neutron applications such as tritium and fissile breeding and electrical power production, as well as radiation effects testing."

This mission leads to significant differences in design approach from that followed in other tokamak test reactor concepts. The FED-R uses normal conducting toroidal-field coils and heavily driven (low- Q_p) plasma operation and requires development of a first wall that is "transparent" to fast neutrons. Other proposed tokamak test reactors, by contrast, are designed to operate at high fusion energy gain, Q_p , utilize superconducting TF coils, and have a broader mission than nuclear testing.

1.2 NUCLEAR TESTING OBJECTIVES

A key feature of magnetic confinement fusion systems is the decoupling of the fusion neutron source from the region of neutron utilization; hence many different types of nuclear tests can be carried out *simultaneously* without affecting the neutron source. Figure 1-1 shows the fluxes and fluences required for the various areas of interest in a test program. The FED-R facility is intended to cover the entire range of interest except that of lifetime radiation damage of structural metals.

The neutronics, thermal hydraulic and materials compatibility studies of Fig. 1-1 would be carried out for a variety of tritium breeding, fissile breeding and electrical-power-producing blanket assemblies. Power production modules would be developed using both "pure fusion" lattices and those containing depleted or natural uranium to provide significant multiplication. Other test assemblies would

Fig. 1-1. Fusion neutron effects program.

include those capable of sustaining exceedingly high temperature ($>1000^{\circ}\text{C}$). This extensive test program is required to validate the technologies of apparently attractive application modules proposed for follow-on fusion facilities.

Two other important nuclear test objectives include the development of large cross section, intense fast neutron beams, which cannot be produced in any other way on a long-pulse basis, and radiation effects studies of electronic devices, insulators and other components including solid breeders expected to be exposed to fast neutron fields.

Some 50 m^2 of first-wall area with 80 cm depth exposed to fusion neutrons can be made available for testing in the region outboard of the plasma vessel, while provisions can be made for altering the neutron albedo of the top, bottom and inboard regions of the vessel. Many large-volume blanket module tests would be carried out simultaneously with radiation effects testing and the production and utilization of fast neutron beams.

1.2.1 Nuclear Application Upgrade

A principal objective of nuclear testing of blanket assemblies in Stage I operation of FED-R is to determine the technical feasibility of numerous technologies for proposed fusion neutron applications. If one fusion application turns out to be in demand for government or commercial purposes in the mid-to-late 1990s, then an option for Stage II operation of FED-R is to replace the vacuum vessel by one located at approximately 0.5-m larger major radius, and install a blanket system that almost completely surrounds the new vessel. The basic performance of this blanket designed for the required fusion application will have been tested in Stage I. This approach can insure that the fusion program is capable of rapidly implementing a practical application of fusion neutrons when one is required.

1.3 NEUTRON WALL LOADING

As suggested in Fig. 1-1, a minimum first-wall flux of the order of $10^{13}\text{ n/cm}^2/\text{s}$ of fusion neutrons, or 0.2 MW/m^2 , is required to set up the

temperature gradients needed for thermo-mechanical and thermal hydraulic studies, including the effects of thermal cycling and realistic convection processes in liquid metals, and also for investigating radiation effects on devices that are dose-rate dependent. An uncollided fluence of 10^{20} n/cm², which is sufficient for lifetime radiation testing of most components and materials other than structural metals, can also be achieved in one year with a 14-MeV flux of 10^{13} n/cm²/s, provided that the reactor operates with a capacity factor of 30% at this level.

To provide more flexibility for tests requiring minimum power fluxes, and to compensate for possibly lower capacity factor than required for some radiation damage studies, a fusion neutron wall loading of 0.4 MW/m² is specified for FED-R in Stage I. The specified wall loading in Stage II, which would be possible in an upgrade of the facility, is 1.3 MW/m². This higher level would permit more rigorous thermal hydraulic studies, radiation effects tests on a wider variety of components that are dose-rate sensitive, and damage rate studies on structural metals if the reactor capacity factor remains high. Even at 0.4 MW/m², thermo-mechanic and thermal hydraulic investigations can be enhanced by the use of depleted uranium in the test assemblies to increase the local power densities.

The minimum electrical power production required for meaningful tests of power module capabilities is probably about 10 MWe. If the blanket energy multiplication is 5, for example, then 10 MWe could be generated using a first-wall area of about one-quarter the available outboard test area when the neutron wall loading is 0.4 MW/m².

1.4 PERFORMANCE OBJECTIVES

In view of the preceding discussion, the performance goals of FED-R have been established as follows:

1. fusion neutron wall loading ≥ 0.4 MW/m² over at least 50 m² test area;
2. fusion energy gain $Q_p \geq 1.0$;

3. accessibility for easy change out of nuclear test modules, loops and other assemblies, of damaged reactor components, and for maintenance;
4. high availability ($\sim 10^7$ burn-s/yr, requiring $\geq 30\%$ availability);
5. steady-state operation, when required, to be achieved with non-inductive current drive; and
6. radiation level of 5 mr/h or less outside of the reactor 36 h after shutdown.

A Q_p of the order of one or larger is required to minimize the capital cost and power requirements of the plasma heating equipment, as well as to ameliorate the heat removal problems in the torus.

For Stage II operation, the performance objectives are upgraded to a fusion neutron wall loading to exceed 1 MW/m^2 at $Q_p \sim 2$ to 3.

Neutral beam injection has been chosen as the primary heating system for FED-R because of its demonstrated success in producing high-temperature, moderate-beta plasmas, and the excellent theoretical prospects as well as experimental indications for neutral-beam current drive. An equally important reason for the neutral beam choice is that if only relatively low values of the confinement parameter $n\tau_E$ can be assured for small test reactors operating at high T_e and β , then neutral beam injection gives the highest probability of reaching and sustaining $Q_p \sim 1$ to 3 because of the large increase in fusion reactivity that results from the injected fast-ion population.

Whether or not sustained high Q_p can be attained in any tokamak reactor, the large injected power needed for steady-state current drive is compatible with the relatively low values of Q_p specified for FED-R.

1.5 FIRST WALL REQUIREMENTS

Test programs to be performed using fusion reactors have been extensively considered and documented.^{a,b,c} The proposed tests span the

^a"The Fusion Engineering Device, Vol. III, Test Plan," DOE/TIC-1160, October 1981.

^b"INTOR Phase One," Section XIV, STI/PUB/619, IAEA, Vienna, 1982.

^cTDF Test Plan.

entire relevant fields of engineering and physics. Only one specific aspect of testing will be discussed here — those tests which are highly influenced by the presence of a first wall/vacuum boundary between the plasma and the test volume. (Section 6.3 contains a more detailed discussion of nuclear testing in FED-R.)

The nuclear test regions in FED-R are located *outside* the vacuum vessel for a number of reasons, including (1) the impact of coolant leakages in test assemblies, (2) the need for rapid removal of test assemblies, and (3) the unusual geometry of some test systems. As a consequence of locating the test modules outside the vacuum boundary, certain tests are more difficult to perform, such as plasma-surface interaction effects (which now require a vacuum port), and first wall radiation tests (except for the FED-R wall). However, most other proposed tests are entirely feasible if the first wall/vacuum boundary is carefully designed.

In the case of certain subgroups of neutronics and radiation effects experiments, the transmission of 14-MeV neutrons into the test module is of concern. Because the test modules are located outside the vacuum boundary, moderation and attenuation of direct-flight (i.e., 14 MeV) neutrons takes place in the wall in front of the test regions. Figure 1-2(a) shows the calculated attenuation of incident 14-MeV neutron radiation as a function of thickness for candidate first-wall materials. Figure 1-2(b) shows the neutron spectrum after transmission through 1 cm of stainless steel, which is approximately the minimum "equivalent" practical wall thickness. (Double walls may be thinner, but may use liquid coolant.)

If the first wall can be made "neutronically thin," then the test modules can be continuously exposed to a bulk fusion source with minimum interference and/or degradation to the fusion spectrum. The term "neutronically thin" is arbitrarily defined in the FED-R application to mean less than 20% attenuation of the 14 MeV neutrons incident on the wall. Several examples for which it is of interest to have small degradation of the high energy neutrons are discussed below.

While this discussion will emphasize the transmission of the 14-MeV fusion neutrons into the nuclear test regions, it is also important

Fig. 1-2. (a) Transmission of neutrons through candidate first-wall materials. (b) Energy distribution of neutrons emerging from 1-cm of stainless steel, 14-MeV source.

that the fast neutron component (0.5 to 5 MeV) that results from scattering in the torus and neighboring components can also penetrate the first wall without undue attenuation.

- (n,2n) reactions

Some blanket concepts depend on (n,2n) reactions in beryllium, lead or other materials to provide enough neutron population for adequate breeding performance. For example, in some tokamak reactor designs the degree of blanket coverage of the plasma chamber is limited to 80% or less, so that the local tritium breeding ratio in the blanket modules must be significantly greater than unity. As another example, the "suppressed-fission" fissile breeding concept depends for its economic viability on maximizing neutron multiplication in beryllium.

The (n,2n) reactions have an effective threshold of about 3 MeV in Be and at least 8 MeV in Pb, Zr, and other candidate metals. Neutronic analyses reveal that 14 MeV neutrons undergoing collisions in the first wall tend to be downscattered by many MeV, so that after a relatively small path length in the blanket module behind the wall their energy will have dropped below the thresholds of interest. Furthermore, many of these neutrons undergoing collision will be scattered to other parts of the torus where their eventual penetration into the blankets will occur at relatively low energy. Figure 1-2(b) shows that even 1 cm of stainless steel will absorb or reflect 9.5% of the incident uncollided current, while approximately 10% of the transmitted current will be insufficiently energetic to excite (n,2n) reactions.

- ${}^7\text{Li}(n,n')$ reaction

Certain tritium breeding blanket concepts obtain vital contributions to their total breeding ratio from this reaction. In a He-cooled Li_2O blanket, for example, 20% or more of the breeding is from the ${}^7\text{Li}(n,n')$ reaction; because of blanket coverage limitations, realizing a significant ${}^7\text{Li}$ reaction rate determines the feasibility of operating without a neutron multiplier. The practical threshold of the ${}^7\text{Li}(n,n')$ reaction

is about 4 MeV. A first-wall thickness much exceeding 1 cm will drastically suppress this reaction.

- Radiation effects

Some of the FED-R test modules would be devoted to radiation damage testing of hardened electronics, insulators, ceramics, and other components that are expected to be exposed to 14 MeV neutrons in various environments. It is important to maximize the 14-MeV neutron component incident on these samples and systems (a) to minimize the residence time needed for lifetime testing; (b) to be able to observe radiation damage caused by neutrons in the 5 to 15 MeV range (the effects of lower-energy neutrons can be studied elsewhere); and (c) to observe transient radiation damage effects on devices and systems that are sensitive to fast-neutron dose rate (rather than total dose). Good transmission of all fast neutrons ($E_n \geq 1/2$ MeV) is important for item (c).

- High temperature regions

Certain proposed fusion applications such as *in situ* "synfuels" production require the maintenance of a very high temperature ($\geq 1,000^\circ\text{C}$) region well isolated from the first wall. Only very energetic neutrons can penetrate adequately through the blanket to sustain this high temperature region.

- Summary

In view of the above discussion, it is desirable to design a first wall and vacuum boundary for FED-R that is neutronically thin, but which is also subjected to acceptable thermal and mechanical stresses and for which the coolant pumping power is reasonable (e.g., 10 MW or less). An integrated approach to achieving this goal appears not to have been addressed previously in fusion reactor design with neutron wall loadings in the range of 1 MW/m² or higher.

1.6 RESISTIVE TOROIDAL FIELD COILS

The requirements that FED-R be capable of deployment by the early 1990s, and that the TF coils be demountable, have resulted in the selection of resistive TF coils for FED-R. By making the coils sufficiently massive, they can be operated steady state at the prescribed magnetic fields with resistive power loss of about 180 MW in Stage I and 280 MW in Stage II, or about one-half of the total facility power requirement.

Demountable joints are included in the upper and lower segments of the TF coils to enhance access for replacement of the in-bore components such as vacuum vessel sectors, poloidal field coils, and the larger nuclear test assemblies. Thus, demountable TF coils permit the objective of good accessibility of in-bore components to be compatible with the cost-driving need for compactness.

Another motivation for demountable TF coils is that if a subsequent stage of reactor operation dedicated to long-term demonstration of some practical fusion application is desired, the horizontal bore of the coils can be extended by means of inserts in the upper and lower legs, and a nearly complete-coverage blanket installed around a new vacuum vessel centered at somewhat larger major radius.

1.7 COMPARISON WITH OTHER NUCLEAR TEST FACILITIES

Mirror machine test reactors have been proposed that ostensibly have a similar mission as FED-R's. The principal differences in performance are a higher wall loading but still lower Q_p for the mirror reactors. Assuming that both tokamak and mirror test reactors are feasible, important differences for nuclear tests are that the first-wall area exposed to fusion neutrons is an order of magnitude smaller in the mirror machines, and the test volume geometry is not capable of accommodating blanket module assemblies that are prototypical of those needed in follow-on reactors.

The advantages of FED-R over the most intense proposed accelerator-drive point-neutron sources (projected 10^{14} - 10^{15} n/cm² at 14 MeV) are the

many orders of magnitude larger test volume irradiated uniformly at fluxes exceeding 10^{13} n/cm²/s, and the ability to vary the ambient neutron spectrum without major loss in neutron intensity.

Fission test reactors have also been proposed for nuclear testing of fusion blanket assemblies. However, the neutron spectrum and power deposition profiles in fission reactor test cavities will be markedly different from those in fusion reactor blanket modules within 30 cm or so from the front face, so that thermal-hydraulic and stress conditions could be notably different. Fission reactor test volumes are also severely limited, and insertion of test samples of significant size into fission reactor cores can change the criticality of the reactor, thereby placing serious new demands on the control rod system. By contrast, in FED-R (or in any tokamak test reactor) the region of neutron utilization is *decoupled* from the neutron source so that test assemblies of arbitrary size and composition can be irradiated. The depth of the outboard test region is nominally 80 cm, and can be made as large as 130 cm.

Intense neutron beams extracted from fission reactors tend to have relatively low energy, and in any event are limited to less than 3 MeV. In FED-R, large cross section, intense beams of 14-MeV neutrons can be obtained for investigation of industrial processes or for nuclear research.

2. DESIGN OVERVIEW

2.1 DESIGN REQUIREMENTS

The requirements set forth in this section are derived from the Mission and Objectives for FED-R discussed in Section 1. Subsequent sections of this report describe how these requirements are satisfied in the FED-R embodiment. In a few cases, however, the requirements have not yet been completely satisfied, so that further design and analysis are needed in these areas.

A two-staged implementation of FED-R is proposed in order to reduce the initial capital investment in device components. The specifications for Stage I call for approximately 3/4 of the magnetic field and plasma current and 1/2 of the injected beam power of Stage II. Consequently, the fusion power and neutron wall loading in Stage I will be just 30 percent of the Stage II values. Requirements are set for Stage II performance levels for those systems which cannot be upgraded between Stage I and Stage II operations. Major parameters of the machine and configurational dimensions are summarized in Appendix A.

2.1.1 Nuclear Test Capability Requirements

1. The fusion neutron wall loading on the outboard vessel wall is to be 0.4 MW/m^2 in Stage I with pulse lengths of at least 1,000 s and duty factor exceeding 50 percent.
2. The region behind at least 50 m^2 of the outboard wall of the vacuum vessel is to be dedicated to nuclear testing.
3. At least 75 percent of the fast neutron current ($E > 0.5 \text{ MeV}$) must penetrate the outboard wall in front of the test regions without scattering. The rationale for this requirement is given in Sect. 1.5.
4. The test region must accommodate test assemblies of arbitrary geometry with a maximum depth of approximately 1.0 m.
5. The annual neutron radiation dose at locations near the front of the nuclear test region is to be at least 10^{12} rads.

6. Provision is to be made to emplace depleted uranium rods or equivalent embodiments in or around test assemblies in order to enhance local heat production and neutron flux. Similar arrangements are to be possible for beryllium or lead components to be used for neutron multiplication.
7. Replacement of test assemblies that can be withdrawn between adjacent TF coil legs should be realizable in a period of 24 h, beginning 36 h after reactor shutdown.

2.1.2 Fusion Plasma Requirements

1. The electron and ion temperatures, energy confinement time, and permissible impurity-ion concentration in the D-T plasma are to be selected at levels necessary to generate a fusion neutron wall loading of 0.4 MW/m^2 at fusion energy gain $Q_p \geq 1$, with credit taken for beam-target and energetic-ion fusion reactions where applicable.
2. The ratio of plasma pressure to magnetic field pressure (β) is to be the maximum value consistent with presently known MHD stability limitations (derived from theoretical analysis buttressed by experimental data where possible).
3. The plasma current must be adequate to support the energy confinement time τ_E and β that satisfy requirements (1) and (2) above, as dictated by present theoretical and empirical understanding of the dependencies of τ_E and β on plasma current.

2.1.3 Selection of Plasma Heating Method

1. The principal method chosen for plasma heating must be a proven technique.
2. The principal plasma heating method must also be capable of driving the plasma current in steady state.
3. Auxiliary heating methods to be used for start-up or profile control should be required to deliver no more than 1/10 of the power of the principal heating method.

2.1.4 Magnetic Requirements

1. The TF coils are to be fabricated of high-conductivity copper.
2. The TF coils are to contain readily demountable joints to facilitate the replacement of in-bore components.
3. The TF coils and normally-conducting PF coils are to be cooled with pressurized water.
4. The shape of the TF coils must be chosen to facilitate test assembly exposure to the fast neutron flux, as well as the installation and removal of nuclear test assemblies.
5. The TF coils must consume no more than 200 MW of power in Stage I operations.
6. The insulation in the TF coils must be capable of sustaining at least 10^{11} rads of nuclear radiation without incurring significant damage.
7. The PF coil configuration, materials, and dimensions are to be chosen for minimal power loss consistent with the need to maintain the equilibrium field and divertor separatrix.
8. The OH (ohmic-heating) solenoid must deliver at least 10 volt-seconds of flux change.
9. The power dissipation in the OH solenoid must not exceed 10 MW during the plasma burn phase (i.e., beginning at approximately 20 s after plasma initiation).

2.1.5 Plasma Start-Up and Operational Requirements

1. Steady-state operating values of plasma current, ion and electron temperatures, and fusion neutron production must be attained by approximately 20 s after initiation of start-up (to insure high duty factor in the event of limited pulse lengths).
2. Dependence on an ohmic-heating solenoid for start-up or current maintenance is to be minimized.
3. Pellet injection is to be used as needed to assist in maintaining the required deuterium and tritium concentrations in the bulk plasma.

2.1.6 Impurity and Particle Control Requirements

1. A single-null poloidal divertor is to accomplish all particle removal and impurity control.
2. All vacuum vessel pumping is to be carried out through the divertor chamber (and neutral beam ducts if applicable).
3. The divertor is to be modularized to facilitate replacement of its hardware components.

2.1.7 Vacuum Vessel and Test Region Requirements

1. The resistance of the vacuum vessel in the toroidal direction must permit full plasma start-up in 20 s, while also being sufficiently low to prevent substantial harm to the vessel in the event of a major plasma disruption, defined as loss of all plasma current in a time of 1 ms or greater.
2. Protection shall be provided against serious erosion of the vacuum vessel wall caused by plasma fluctuations or neutral beam shine-through.
3. The vessel coolant system must be capable of removing a steady-state thermal loading of 60 W/cm².
4. The vessel wall with ancillary structures is to be neutronically thin in front of the nuclear test regions. For example, at least 75 percent of the fast neutron current ($E > 0.5$ MeV) should pass unattenuated into the test regions. The rationale for this requirement is given in Sect. 1.5.
5. Nuclear test assemblies are to be replaceable by overhead crane, with replacement by radial translation an option to be provided where practicable.
6. Shielding behind the inboard wall must limit the lifetime radiation dose of the TF coil insulation to 10^{11} rads.
7. Provisions are to be made for changing materials in a region at least 20 cm deep behind the first wall on the inboard and top of the vessel, as required to vary the intensity and energy spectrum of neutrons scattered to the outboard side.

8. Outboard and top shielding sections must be segmented to simplify removal of test assemblies.

2.1.8 Availability and Maintenance Requirements

1. The availability of the FED-R subsystems must be sufficiently high to permit an annual capacity factor of at least 30 percent in Stage I and Stage II.
2. The radiation dose level at the TF coil joints must be sufficiently low at 36 h after shutdown to permit partial hands-on disassembly of the joints.
3. An outboard TF coil leg must be capable of disassembly and reassembly by operations requiring a total elapsed time of no more than one week.
4. A damaged TF coil must be replaceable in no more than two months.
5. Divertor modules must be removable without disturbing the TF or PF coils.
6. Access to the vacuum vessel sector joints and duct interfaces must be adequate to permit rapid leak detection and repair.
7. A damaged vacuum vessel sector must be replaceable in no more than two months.
8. Personnel access to the outside of the reactor must be permissible at 36 h after shutdown.
9. NBI ion sources must be readily accessible for remote replacement.

2.2 DESIGN APPROACH

2.2.1 Use of Near-Term Technology

The main goal of FED-R is to provide a large test area exposed to fusion neutrons for the purpose of nuclear engineering testing, including the development of breeding blanket concepts and other fusion neutron applications. Achieving this goal in an early-1990s test reactor demands the use of current or mid-1980s fusion technologies. For example, normal-conducting water-cooled toroidal field TF coils have been selected

for FED-R. These coils are constructed from plate segments which can be factory-built and shipped to the reactor site for assembly.

The objective of maximum test area without violating the vacuum boundary dictates that the vacuum vessel wall be located next to the plasma. Activation of components, the presence of tritium, and the general complexity of the electromagnetic features of the tokamak device significantly influence the maintenance and repair operations. The FED-R maintenance approach has led to a modularized design concept. Designing to achieve the required access has had a significant impact on the configuration of the tokamak systems. The most notable areas include 1) selection of normal toroidal field coils, 2) selecting a hybrid PF system configuration (some coils inside the TF coil bore and some coils outside) over an all-exterior configuration, and 3) minimizing the overall device size.

The reference bulk heating technique in FED-R is the proven one of neutral beam injection. For Stage I operation in the early 1990s, the provision of 150-keV D^+ -based neutral beams is specified, as such beams represent in most respects a relatively modest step beyond the TFTR beams. However, a major extrapolation is that the beams must operate steady state; the development of 80-keV, 30-s beams for MFTF will provide important developmental steps in this regard.

2.2.2 Plasma Physics Basis

Because the principal goal of FED-R is to provide a reliable fusion neutron testing capability, both the plasma physics and technological features of the FED-R device must be chosen as close to mid-1980s state of the art as possible. Hence the plasma parameters (T_e , T_i , β , $n\tau_E$) specified for Stage I operation should be near those expected in TFTR or D-III Upgrade.

Confinement. Possible obstacles to reaching high τ_E are strongly indicated by present high-powered beam and RF-heated tokamak confinement experiments. If only relatively low values of $n\tau_E$ can be assured for small test reactors operating at high values of plasma temperature and near the MHD-limiting beta, then neutral-beam injection (NBI) offers the

greatest potential for enhancing fusion gain Q_p because of the increase in overall fusion reactivity resulting from a large population of injected energetic ions. For example, $Q_p \geq 1$ can be obtained even for $\bar{n}n_E$ as low as $1 \times 10^{13} \text{ cm}^{-3}$ by taking advantage of beam-target and beam-beam reactions. The prospect of relatively low n_E , at least in quasi-steady-state operation, is why small values of Q_p are specified for FED-R, and is also one reason why NBI is the preferred heating technique.

Beta. Present experimental results from ISX-B, PDX, and D-III indicate that the maximum spatially averaged beta, $\langle\beta\rangle$, achievable in beam-heated circular tokamak plasmas is about 3%, which is in reasonable agreement with the marginal stability limits predicted by MHD theory. The same theory predicts that D-shaped plasmas are capable of attaining $\langle\beta\rangle$ up to about 6% for vertical elongations in the range 1.6 to 2, aspect ratios of 3 to 4, and $q_\psi \sim 2$. Recent results from D-III showing $\langle\beta\rangle$ up to 4.5% at $\kappa \approx 1.5$ provide strong evidence for the theoretical enhancement of beta in elongated plasmas.

The $\langle\beta\rangle$ required in FED-R is approximately 5.0%. It is assumed that this value can be attained for $\kappa = 1.5$ to 1.6, $R/a = 4$, and $q_\psi = 2.25$. Slightly lower values of q_ψ may actually be required, but this possibility must await the results of further tokamak experiments.

There are also some experimental and theoretical indications that higher values of β are achievable in plasmas where a significant fraction of the pressure is in the form of energetic streaming ions. This result, if valid, would be important in low-Q devices that use quasi-tangential neutral-beam injection, and is another reason for specifying NBI heating for FED-R.

High duty factor. The one vital difference in the required plasma operation of FED-R compared with the operational requirements of TFTR and D-III Upgrade is the quasi-continuous operation demanded for FED-R. High-duty-factor operation implies closely repetitive pulses of at least 500 s in length, and steady-state plasmas would be preferable.

To achieve the highest values of $\langle\beta\rangle$, plasma currents of the order of 5 MA are required in near-term machines with plasma radius ~ 1 m and $B_t \sim 4$ to 5 T. To generate these currents with noninductive current drive, as needed for steady-state operation, will require at least 50 MW

of injected power. This large power is compatible with the low- Q_p operation specified for FED-R.

To date definitive demonstrations of current drive have been made only with lower hybrid waves (LHH), although there exists significant evidence of neutral-beam-driven current in the DITE tokamak, and some indications from ORMAK and PLT. On the other hand, LHH has been relatively ineffective in plasma heating. Ion cyclotron waves (ICRH) have proved to be very effective in plasma heating, but there is no evidence for current drive with ICRH and even its theoretical prospects are uncertain. The undeniable success of NBI in producing high-temperature, moderate-beta plasmas, as well as the existence of some evidence for neutral-beam current drive, provides the most important rationale for choosing NBI for the bulk heating of FED-R plasmas.

Particle control. High-duty-factor operation also requires the continuous removal of large fluxes of thermal energy and particles diffusing out of the discharge as well as disposing of sputtered impurity ions. Experiments in PDX, ASDEX, and various smaller tokamaks have demonstrated that these functions can be achieved with a poloidal magnetic divertor. Because of the uncertainties in the use of pumped limiters for extended operation, a poloidal divertor has been specified for FED-R.

Experiments in the ORMAK, ISX-B, and PLT tokamaks have shown that injection of neutral beams in the direction of the plasma current ("co-injection") is effective for impeding the flow of impurities to the center of the plasma. This "flow reversal" effect is also predicted by neoclassical plasma transport theory. The co-injected neutral beams planned for FED-R should assist with the diffusion of charged fusion-reaction products (mainly D-T alphas) to the edge of the plasma where they can be captured eventually by the magnetic divertor.

Start-up. Analysis shows that in FED-R the entire plasma current, I_p , can be sustained by the planned neutral-beam injection (50 MW in Stage I and 100 MW in Stage II). Consequently, I_p can be started up with neutral beams assisted by part of the flux swing associated with the EF system. The lower T_e that exists during most of the current

start-up phase, tends to reduce the beam-driven current, but the higher Z_{eff} during start-up enhances current drive. In order for the desired operating value of I_p to be reached, neutral-beam heating must simultaneously increase T_e to its operating value during current ramp-up.

A plasma current of about 300 kA and a density of $2 \times 10^{13} \text{ cm}^{-3}$ are required in the initial target plasma to insure neutral beam stopping and fast-ion capture. Experiments on ISX-B and Soviet tokamaks have demonstrated that electron cyclotron radiation can generate plasmas with T_e of some tens of volts, and substantially higher T_e in the resonant layer. This level of T_e reduces the initiation loop voltage to about 15 V. The total flux swing required of the OH system in order that it can establish the initial current of 300 kA is then 10 to 15 Wb, depending on impurity content. The OH solenoid for FED-R is specified to have a capacity in this range.

2.3 DESIGN DESCRIPTION

2.3.1 Configuration

Figures 2-1 and 2-2 present the FED-R configuration in elevation and plan views for Stage I operations. Figure 2-3 presents the Stage II configuration for FED-R. The main difference consists of replacing the three D^+ -based neutral beams with three higher energy D^- -based ion beams (same injection angle) and adding a T^- -based neutral beam with a shallower injection angle. Table 2-1 shows a comparison between FED-R Stage I and TFTR parameters.

The FED-R maintenance approach requires (1) modularity of components (such as divertor sectors) which are expected to require periodic replacement or maintenance, (2) accessibility of test modules and other components placed inside the TF coil bore, and (3) hands-on access to all components outside the shield 36 h after shutdown.

2.3.2 Magnetic System

The FED-R magnetic system consists of 12 essentially rectangular toroidal-field (TF) coils, six equilibrium-field (EF) coils, and a

TYPICAL CROSS SECTION

Fig. 2-1. FED-R Elevation - Stage I.

ORNL-DWG 82-3005 FED

Fig. 2-2. FED-R Plan - Stage I.

ORNL-DWG 82-3006 FED

Fig. 2-3. FED-R configuration - Stage II.

Table 2-1. Comparison of principal parameters of FED-R
Stage I and of the TFTR (on line in 1983)

Parameter	FED-R (Stage I)	TFTR
Major radius	3.5 m	2.5 m
Minor radius	0.85 m $\times$ 1.30 m	0.85 m
Maximum field at coil windings	7.0 T	9.2 T
Maximum field on plasma axis	4.0 T	5.2 T
Plasma current	3.5 MA	2.5 MA
Safety factor at plasma edge	2.25	3
Neutral beam energy	150 keV	120 keV
Neutral beam power	50 MW	35 MW*
$\bar{n} \tau_E$	1 to 3 $\times 10^{13}$ cm ⁻³	$\geq 1 \times 10^{13}$ cm ⁻³ s
β (Bulk plasma + fast ions)	5 to 6%	2.5 to 3.5%
Pulse length	10,000 s	1 s
Duty factor	95 to 100%	0.3%
Fusion energy gain	1.5	~ 1
Neutron wall loading at outboard midplane	0.4 MW/m ²	0.15 to 0.3 MW/m ²
Useful neutron test area	60 m ²	1 to 4 m ²

* Including fractional energy neutrals.

central ohmic-heating (OH) solenoid. The TF coils, OH solenoid, and the internal poloidal-field (EF) coils employ a copper conductor cooled with water. Resistive TF coils are used because of the need for demountable joints. However, two external PF coils (large radius ring coils) are superconducting in order to reduce substantial power losses.

The configuration of TF and PF coils is shown in Fig. 2-1. The TF coils are designed to provide up to 5.0 T on the plasma axis, although only 4.0 T is required for Stage I. In order to achieve the specified on-axis fields, the TF coils are designed to develop 7-T and 8.8-T peak field at the TF coil windings for Stage I and II, respectively. The ohmic heating (OH) coil is designed to operate at fields up to 9.5 T; this magnet employs a Bitter-plate winding cooled with water.

The TF coils employ a copper conductor in the form of water-cooled plates with a bore of 3.7 m x 7.2 m. Each coil has 17 turns, each of 6-cm thickness, with the coolant water flowing through tubes placed in milled slots. The plates are insulated from each other by ceramic insulation (SPINEL) applied by plasma arc spraying. There are 3 types of joints in the coil: demountable finger joints, permanent manufacturing lap joints, and electrical joints for the turn-to-turn transitions.

2.3.3 Torus System

Vacuum Vessel

The vacuum vessel is located next to the plasma and consists of six vacuum vessel sectors and six internal shield frames which protect the seal welds from direct plasma exposure. The vacuum vessel is also the first wall and contains the armor and poloidal divertor, all of which must operate in a high neutron flux environment and are subjected to potential plasma disruptions. The vacuum vessel wall in front of the test modules is cooled with helium to provide maximum neutronic transparency. To provide for ready access to these components, openings are placed in the vacuum vessel for in situ armor tile inspection and replacement. The potential for damage from thermal and particle loads on the poloidal divertor is high, and hence provisions are included in the

design for frequent removal and replacement of the divertor module, if required. The divertor is divided into twelve modules, the same number as the TF coils, to allow for module removal without disturbing the TF coils. Replacement of a vacuum vessel sector requires the prior removal of the outboard legs of 3 adjacent TF coils.

Bulk shielding

The FED-R bulk shielding is located totally outside the vacuum boundary. The shield design uses layers of Nitronic 33 and water in specified thicknesses and locations. The shield is actively water cooled. Each bulk shield component is treated as a pressure vessel with an independent water coolant system.

Divertor

The divertor collector system undergoes the most severe thermal stress and corrosion of all components in the torus, and the divertor design includes provisions for frequent replacement, if necessary. Modular divertor sectors can be removed by radial extraction. Each set of divertor modules is provided with an exhaust duct from which it can be disconnected without dismantling the pumping system. The collector plates and the other divertor surfaces subject to potential erosion are protected by tiles of refractory metal attached to the surface and are water cooled.

2.3.4 Electrical Systems

Plasma Heating

The primary heating sources are steady-state neutral beam injectors. For Stage I, three D^+ -based neutral beams provide 50 MW of injected power. For Stage II, four D^+ -based neutral beams provide 100 MW of injected power. Two ECRH waveguides provide a total of 2 MW of rf power for start-up and control heating.

Power handling and conversion

The electrical power handling and conversion system includes the ac power system and the TF and PF coil power conversion systems. The ac power system provides both pulsed and steady-state power for the FED-R loads. The required ac power system capacity is 500 MVA for pulsed power loads. The TF coil power conversion provides for charging the twelve TF coils (in about 600 s with two 65 V power supplies). The PF coil power convertors take ac power from the motor-generator-flywheel (MGF) units (or utility line) and convert it to pulsed dc power needed for the PF coils during each operating cycle.

Electrical energy storage

The electrical energy storage requirements are met in FED-R with a minimum of two MGF units. Energy storage is required for the PF coils and for the rf systems. A total of 6.4 GJ of energy is required during startup with a peak MVA load of 600. The motor-generator-flywheel (MGF) units satisfy these requirements. Each MGF unit has a wound-rotor induction motor of 15,000 HP. These units are safe, reliable, economical, and easy to control. Voltage can be regulated to within $\pm 1\%$ with conventional controls.

Fueling systems

The FED-R fueling system consists of gas puffers and pellet injectors, as well as the fueling provided by the neutral beam injectors. Two independent gas puffing systems (one for redundancy) are available to provide fuel gas (deuterium, tritium, or a mixed species) to each of 12 inlet ports. The gas puffing is used to backfill the torus prior to start-up and continues until a plasma temperature of 1 keV is established. Frozen deuterium and tritium pellets are then injected to control plasma density. Two pellet fuel injectors are located in one torus sector. Each can inject 4-mm pellets at a velocity of 2 km/s at an injection rate of up to 20 pellet/s.

2.3.5 Facilities Description

The FED-R facility includes the reactor building, hot cell facility, necessary additional support buildings, and a site layout.

The reactor building is a rectangular building with dimensions of $64 \times 57 \times 36$ m. The walls and roof are 2-m thick to provide for adequate shielding. The building is designed to handle postulated accident conditions. The hot cell facility provides the capability to support the maintenance and operation of the reactor building and those other facilities involving radioactive operations. The hot cell facility has dimensions $78 \times 52 \times 32$ m and is of a controlled ventilation construction. Walls and roof construction are of 2-m-thick concrete. The additional facilities required for the total FED-R complex are identified in Sect. 8.3.

2.4 DEVICE OPERATION

This section qualitatively summarizes the operation of a typical FED-R pulse in either Stage I or Stage II.

A line-averaged density of at least $2 \times 10^{13} \text{ cm}^{-3}$ is needed to trap the injected beam neutrals and a plasma current of at least 300 kA is needed to provide adequate orbit confinement of the energetic ions formed from the beams. This density and current are produced in less than 2 s, by first preionizing the fill gas and heating the electrons to 10 to 50 eV with 2 MW of ECRH, then initiating the ohmic-heating transformer to generate a 300 kA current at $T_e \sim 500$ eV, while raising the density to $2 \times 10^{13} \text{ cm}^{-3}$. The neutral beams are then activated. In combination with the considerable flux swing provided by the equilibrium-field coils, the increasing neutral beam power takes the plasma current to its final value in 15 to 20 s. Simultaneously, the beams heat the plasma electrons and ions to their operating temperatures.

The injected beams maintain the plasma current and temperature throughout the burn pulse, with some changes in beam power, slight changes in beam voltage, and varying pellet injection rates in order to keep the fusion neutron output at the prescribed level for that pulse.

While an almost indefinitely long pulse is possible in principle, the pulse length is specified as 3 h in Stage I and 8 h (1 shift) in Stage II.

Fueling by the deuterium neutral beams is sufficiently intense so that in the center of the plasma column, the deuteron density cannot be balanced by injection of tritium pellets of acceptable size. Hence one of the neutral beam injectors is used to inject tritium beams. Particle pumping and impurity control are accomplished by the single-null poloidal divertor, which also remove approximately 75 percent of the non-nuclear power flux from the plasma.

The location of the nuclear test regions outside the vacuum vessel wall allows rapid changeout of the test assemblies regardless of their geometry (cubical modules, vertical rods, vertically oriented ducts, liquid metal loops, neutron beam tubes, etc.). Averaged over the year, there will be just one shift of operation per day. One or more test assemblies can be replaced after each shift while maintaining the torus vacuum. Replacement would be made frequently of those test assemblies installed only for neutronic, thermal hydraulic, materials compatibility, or radiation effects studies with dose-rate-dependent components. If removal of a particular assembly cannot be accomplished remotely, then a waiting time of 36 h will be required.

Test assemblies more than 1.6 m in width would require disassembly of an outboard TF coil leg for removal or installation. That operation would take up to two weeks round-trip, and would be made only in conjunction with significant maintenance activities such as the replacement of divertor modules or the repair of in-bore coolant systems.

2.5 COST AND CONSTRUCTION SCHEDULE PROJECTION

2.5.1 Cost Estimate

As previously noted, the FED-R project employs a two-stage approach to nuclear operation which entails: a three-year prenuclear test phase; a six-year Stage I test period using low (0.4 MW/m^2) wall loading; a 1.5-year upgrade period; and a 9.5-year Stage II test period at high

(1.3 MW/m²) wall loading. In preparing the Stage I cost projection, only the required complement of power supplies, heat transport equipment, and plasma heating equipment necessary for Stage I operation were included in the Stage I costs. Stage I *direct* cost is projected to be \$810 million.

The increment of cost required to upgrade FED-R to the Stage II configuration covers the additional power supplies, energy storage equipment, heat transport equipment, and additional plasma heating equipment necessary for Stage II operation. The total capital cost increment required for the Stage II upgrade amounts to \$220 M, for a total *direct* cost of \$1030 M.

Inclusion of indirect costs (engineering and management, installation, and contingency) increases the total estimated cost to \$1686 M in Stage I and \$2145 M in Stage II.

2.5.2 Construction Schedule

A preliminary projection of the FED-R construction schedule has been developed (Section 10). The approach taken was to use the FED baseline schedule⁴ as a model. Initial project startup activities for FED-R were judged to be the same as those of the baseline and with a required time span of 24 months. The FED-R reactor building (which is the critical path facility construction item) was found to be about 10 percent smaller than the baseline. Floor area increases may be required to accommodate copper TF coil cooling equipment. However, it was considered reasonable to use the baseline time span of 21 months for FED-R reactor building construction.

A revised component installation sequence was developed for FED-R, since the basic design approach and hence the order of component installation represented a departure from the FED baseline. Installation time spans for FED-R components were determined by comparison of complexity relative to their FED baseline counterparts. Adjustment of the time spans were made accordingly. Total time from start of FED-R equipment installation to start of integrated system test is 35 months. The time

allocated for FED-R integrated system checkout is projected to be 12 months.

The total design, fabrication, and assembly schedule for FED-R is projected to be of 92 months duration. The 92 month figure is considered to be a lower bound, since no allowance has been made for contingencies or changes. Further details regarding the FED-R schedule are in Sect. 10.2.

2.6 FUTURE DESIGN WORK

2.6.1 Higher Beta Plasmas

Higher plasma current

Recent tokamak experiments using intense neutral-beam injection show that both the maximum attainable $\langle\beta\rangle$ and the energy confinement time increase significantly as the plasma current I_p is raised at constant field (i.e., as q_ψ is lowered). Hence the advantages of a somewhat higher I_p in FED-R should be assessed. For example, the reference currents of 3.5 MA in Stage I and 4.4 MA in Stage II correspond to $q_\psi = 2.25$, which could be reduced by 10 to 20%. This requirement would put greater demands on the OH system for startup and on the neutral beams for current drive.

Greater plasma elongation

MHD stability analyses indicate that for tokamak plasmas with aspect ratios of about 4 (such as the FED-R case), maximum $\langle\beta\rangle$ is attained for plasma vertical elongation κ of about 1.8, rather than the 1.5 to 1.6 used in the reference design. The triangularity of the plasma should probably be increased also from its present value of 1.25 to about 1.5, to attain maximum $\langle\beta\rangle$.

An increase in κ to 1.75 can be implemented in FED-R within the present bore of the TF coils by reducing the top and bottom shielding by 15 cm. The SPINEL insulation in the TF coils will be able to withstand the increased dose. The vertical height of the plasma would then

increase to about 1.50 m. The 12% increase in vertical elongation would require approximately a 15% increase in I_p to maintain the same q_ψ , which would put further demands on the OH system and neutral beam current drive.

Present experiments on Doublet III and elsewhere indicate that the higher current associated with the increase in plasma elongation will also result in an increase in $n\tau_E$, giving a higher Q_p and neutron wall loading for the same injected beam power.

(An incidental aspect of higher κ is that taller nuclear test assemblies would be permitted. In the case of sectors holding neutral beam ducts, the change would allow modules of more significant size to be located on top of the ducts.)

Bean-shaped plasmas

The reasonable success of MHD equilibrium and stability analyses in predicting the approximate beta limitations of circular and D-shaped tokamak plasmas (ISX, PDX, Doublet III) has led to renewed interest in the theoretical performance of higher elongation plasmas with triangularity of the order of unity. Extensive theoretical work has been initiated recently on a class of plasmas of bean-like cross-sectional shape. It appears that $\langle\beta\rangle$ -values in the range of 8 to 12% could be obtained with an elongation of about 2.5 with moderate "bean shape."

The feasibility of achieving $\langle\beta\rangle \sim 10\%$ could be greatly advantageous to the FED-R, inasmuch as the TF coil power dissipation and the TF coil stresses would be roughly halved. The reactor implications of high- β bean-shaped plasmas have not yet been addressed. A worthwhile future study is to investigate, in the context of FED-R, issues such as the requirements for the PF coil configuration and currents, methods of plasma formation and maintenance of its current profile, and the interface with a poloidal divertor.

2.6.2 Ohmic Heating System

If neutral-beam current drive turns out to be partially or totally inadequate for assistance in plasma current start-up, or for driving the

FED-R current in steady state, then fallback schemes involving the OH system must be devised. Furthermore, the beta-enhancement methods discussed in Sect. 2.6.1 require higher I_p than the reference FED-R values, placing greater demands on the OH system or the neutral-beam current drive. Hence it appears worthwhile to carry out investigations on approaches to increasing the capability of the OH solenoid. This subsection outlines such approaches.

The two basic steps are the following:

1. Increase the central throat of the machine, without changing the plasma dimensions by reducing the inboard radiation shielding.
2. Increase the flexibility of the throat for acceptance of a solenoid that can be either water-cooled, liquid-nitrogen-cooled, or superconducting.

Increased diameter of throat

The throat diameter can be increased while moving the inboard legs of the TF coils outward so that the coil power dissipation is no greater than in the reference case. For example, an increase in the throat radius from 72 cm to 80 cm results in the same TF coil power loss if the inboard legs are moved outward by 5 cm. To keep the plasma major radius constant, the thickness of the inboard coil shielding would be reduced from 45 cm to 40 cm. The consequent increase in the lifetime radiation dose at the TF coil insulation to 1×10^{11} rads is still acceptable for the SPINEL insulator.

(For the same field at the plasma, the maximum field at the coil would be reduced by 2.5%, and the coil stress reduced by 5%.)

Because the SPINEL magnet insulation is thought to be capable of withstanding 10^{12} rads before significant damage occurs, another 20 cm of shielding could be removed and the TF coil legs moved outward by this amount, so that the throat radius would be increased to 1.0 m. One restriction on these changes in component locations might be the amount of nuclear power that would have to be removed by the TF coils.

An increase in the throat radius from 7 cm to 100 cm would increase the OH flux swing from 11 Wb to at least 20 Wb at constant B_{OH} .

Increased flexibility

Except for their common gravity support, the OH solenoid in the reference design is structurally independent of the TF coils. With a larger throat diameter, this physical decoupling could be enhanced. Completely independent structural support of the solenoid would enable a water-cooled system to be replaced by a liquid-nitrogen (LN)-cooled system, or by a superconducting solenoid.

LN-cooled solenoid

With an OH-solenoid of the Alcator type cooled to 77 K at the beginning of the pulse, much higher B_{OH} and hence flux swing are possible because of the small resistive power loss at cryogenic temperature. (The actual energy requirement is very large because of the power needed to refrigerate the liquid nitrogen. However, this energy is expended at low power during the lengthy burn.) A typical design of the solenoid would have 100 pancake coils with 10 turns per coil. The maximum current density would be 4 to 5 kA/cm², B_{OH} would be as large as 18 T, and at least 35 Wb could be obtained in a double swing. (As in the reference design, the solenoid current would be returned to zero during the first 100 s of the burn, with the neutral beam current drive counteracting the reverse EMF.)

While inertial cooling would be simplest, hollow conductors could be used to carry the LN in forced flow. The same pancake coils could in fact be used either with water cooling or LN cooling.

Superconducting solenoid

An optional method of plasma current start-up is to apply 5 to 10 MW of lower hybrid radiation at low density (a demonstrated technique) and then use a superconducting coil solenoid to maintain the plasma current. A pulse length of 600 s would be possible with a typical loop voltage of 0.025 V and a flux swing of 15 Wb. The solenoid would have to be superconducting, in order to eliminate unacceptable resistive

power loss. In this application, the maximum rate of change of B_{OH} would be less than 1 T/s, which is thought to be within the state of the art. The feasibility of this procedure would seem to be assured on the basis of present-day experiments and technology, but it would be more costly than the reference method of using beam current drive (inasmuch as the beams are injected in any event for plasma heating), and steady-state operation would not be possible. Nevertheless, it provides a viable back-up scheme in the event of shortcomings of beam current drive.

REFERENCES

1. D. L. Jassby, *TORFA - Toroidal Reactor for Fusion Applications*, PPPL-1700, Plasma Physics Laboratory, Princeton University (1980).
2. D. L. Jassby, R. A. Jacobsen, J. Kalnavarns, L. S. Masson, J. Sekot, "Resistive Demountable Toroidal-Field Coils for Tokamak Reactors," in *Proc. 4th Topical Meeting on the Technology of Controlled Nuclear Fusion* (King of Prussia, Pa., 1980), PPPL-1809, Plasma Physics Laboratory, Princeton University (1981).
3. J. Kalnavarns and D. L. Jassby, *Resistive Toroidal-Field Coils for Tokamak Reactors*, PPPL-1685, Plasma Physics Laboratory, Princeton University (1980).
4. *Fusion Engineering Device*, DOE/TIC-11600, October 1981.
5. C. A. Flanagan, D. Steiner, and G. E. Smith, *Fusion Engineering Device Design Description*, ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee, December 1981.
6. R. J. Fonck et al., "Impurity Levels and Power Loadings in the PDX Tokamak," in *Proc. 5th Int. Conf. on Plasma Surface Interactions in Controlled Fusion Devices* (Gatlinburg, Tenn., May 1982), to be published; the ASDEX Team, "Neutral Beam Heating of ASDEX Divertor Plasmas," in *Proc. 9th Int. Conf. on Plasma Physics and Controlled Fusion Research* (Baltimore, Md., Sept. 1982), paper A-3, to be published by IAEA, Vienna.
7. P. N. Haubenreich et al., "Superconducting Magnets for Fusion." *Nucl. Fusion* 22 (1982) 1209-36.
8. A. M. M. Todd et al., "Dependence of Ideal MHD Kink and Ballooning Modes on Plasma Shape and Profiles in Tokamaks," *Nucl. Fusion* 19 (1979) 743-52.
9. R. M. Gilgenbach et al., "Electron Cyclotron/Upper Hybrid Resonant Pre-Ionization in the ISX-B Tokamak," *Nucl. Fusion* 21 (1981) 319-327.
10. *Proc. 9th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research* (Baltimore, Md., Sept. 1982), papers A-1 through A-5, to be published by IAEA, Vienna.

11. M. Nagami et al., "High-Beta Injection Experiments with Shaped Plasmas in Doublet III," in *Proc. 9th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research* (Baltimore, Md., Sept. 1982), paper A-2, to be published by IAEA, Vienna.
12. D. J. Grove, "The Status of TFTR," in *Proc. Fourth Topical Meeting on the Technology of Controlled Nuclear Fusion*, CONF-801011, U.S. Dept. of Energy (1981), pp. 957-60.
13. L. G. Davis and J. M. Rawls, "Modification of Doublet III," in *Proc. 9th Symp. on Engineering Problems of Fusion Research* (IEEE, Chicago, 1981), pp. 1939-42.

3. CONFIGURATION DESIGN

The system design includes configuration design and integration, assembly and maintenance, and system trade studies. A brief summary of the device configuration requirements and sub-system design is provided in this section. The impact of the machine assembly/disassembly/maintenance requirements on the configuration design is addressed.

3.1 REQUIREMENTS

The main goal of FED-R is to provide a test bed for nuclear assemblies such as breeding blanket modules. It is required that the test assemblies should be readily accessible. Considerations that have influenced the evolution of the FED-R configuration are: 1) the required operating parameters, 2) the objective of maximizing the test module area, 3) the maintenance criteria, and 4) the capital and operating costs. It was found that physics considerations impact the device configuration primarily in the particle and impurity control concept and in the PF system concept. On the other hand, the objective of maximum test area and replacement of test modules without violating the vacuum boundary dictated the location and placement of auxiliary equipment.

Activation of components, the presence of tritium, and the general complexity of the electromagnetic features of the tokamak device significantly influence the maintenance and repair operations. Accordingly, maintenance considerations were established at the outset of the FED-R design study as a fundamental consideration in the development of the design configuration. Implementation of the FED-R maintenance approach (see Sect. 9) has led to a modularized design concept, and designing to achieve the required access has had a significant impact on the design of the tokamak systems. The most notable areas include demountability of the normal toroidal field (TF) coils and selection of a hybrid PF system configuration (with some coils inside the TF coil bore and some coils outside the bore) over an all-interior PF system configuration.

3.1.1 Availability

The availability* which FED-R is capable of achieving is determined by the reliability, maintainability, and supportability characteristics of the design and by scheduled maintenance requirements.

A plan is proposed which integrates FED-R testing requirements into sequenced phases of operation and establishes guidelines for operation in each phase. Table 3-1 summarizes the phases of operation, operating guidelines, and the estimated number of pulses associated with each phase. The availability shown is that associated with accomplishing the indicated number of pulses in the time allotted.

The lower level of availability requirements for the earlier phases is consistent with operational constraints which are expected to impact initial FED-R operation. These operational constraints include:

- Hardware checkout and software debugging
- "Infant mortality" failures due to deficiencies in initial design, fabrication or assembly
- Initial unfamiliarity with maintenance procedures
- Incorporation of design upgrades for improved performance

The availability requirement is most severe in the DT phases of operation when FED-R must demonstrate a capacity factor of at least 32%. Nominal operation will be for five days per week, twenty-four hours (three shifts) per day. (Maintenance crews are assumed available around-the-clock, seven days per week.) A three week period every third month has been identified for scheduled maintenance and reconfiguration activities. FED-R would have to operate 59% of the scheduled operating time in order to achieve a capacity factor of 32%.

In developing the overall availability requirement, it is important to recognize global maintenance constraints including:

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* Availability is defined as the ratio of operating time to operating time plus downtime. For the purpose of this analysis, all nonoperating periods are assumed to be downtime periods wherein maintenance would be accomplished.

Table 3.1. Proposed plan of FED-R operation

Phase	Duration (Years)	Description	Operating Guidelines	Number of Pulses ^a	$\frac{SOT^b}{CT}$	$\frac{OT^c}{SOT}$	Availability ^d
A	0.5	Integrated systems checkout	2 shifts/day 5 days/week 2 weeks/4 weeks				
B	1.5	Hydrogen deuterium operation	2 shifts/day 5 days/week 4 weeks/6 weeks	5,000	0.32	0.50	0.16
C	1	Tritium systems check-out	As per phase B				
Stage I	6	Low wall loading DT operation	3 shifts/day 5 days/week 10 weeks/13 weeks	6,000	0.54	0.59	0.32
D	1.5	Equipment upgrade	—				
Stage II	9.5	High wall loading DT operation	3 shifts/day 5 days/week 10 weeks/13 weeks	3,325	0.54	0.59	0.32

^aAverage pulse length: Phase A - 100 s, Phases B and C - 2,000 s, Stage I - 10,000 s, Stage II - 28,000 s

^bScheduled operating time/calendar time

^cOperating time/scheduled operating time

^dOperating time/calendar time

1. A 36-hour cooldown time is required prior to manned access to the reactor cell.
2. Sixty-four hours are required for detritiation prior to letting the plasma vacuum chamber up to air.
3. One week is required for reconditioning the plasma vacuum chamber once it has been let up to air.

Item nos. 2 and 3 indicate one reason why the nuclear test assemblies, which are to be replaced frequently, are located *outside* the vacuum vessel wall.

The operating guidelines for the DT phases of operation thus allow scheduled manned access to the reactor cell for twelve hours per week. During this time, scheduled changeout of test modules can be performed. Scheduled maintenance that requires letting the plasma vacuum chamber up to air, e.g., changing diagnostics and divertor plates, as well as large scale replacement of neutral beam sources must be accomplished in a coordinated fashion during the maintenance windows which occur at three month intervals. Although the device is scheduled to be down for 23 days during these maintenance windows, approximately 10 days will be required for plasma chamber detritiation and reconditioning alone if the plasma chamber is let up to air.

3.2 DESIGN DESCRIPTION

Figures 3-1 and 3-2 present the FED-R configuration in an elevation and plan view for Stage I operations. Figure 3-3 presents the Stage II configuration for FED-R. The main differences between the Stage I and Stage II configurations consist of replacing the three positive ion-based neutral beams with three negative ion-based deuterium beams (same injection angle) and adding a negative ion-tritium beam with a shallower injection angle. (In an optional Stage II upgrade outlined in Appendix C, the TF coil bore is enlarged by means of inserts in the upper and lower coil arms.)

The magnetic system consists of 12 TF coils, six EF coils, and a central solenoid. The TF coils are constructed from water-cooled copper

TYPICAL CROSS SECTION

Fig. 3-1. FED-R elevation - Stage I.

ORNL-DWG 82-3005   FED

Fig. 3-2. FED-R plan - Stage I.

Fig. 2-3. FED-R configuration - Stage II.

plates. Two of the EF coils are the superconducting type. The design basis is the same as that employed for the FED baseline.^{1,2} The ohmic heating central solenoid employs a Bitter-plate winding which is cooled with water.

The torus vacuum vessel is divided into six 60-degree sectors for ease of construction. The vacuum wall is placed next to the plasma. The bulk shielding is located outside the vacuum boundary. The vacuum vessel is cooled with water except in front of the test modules, where helium coolant is used to provide maximum neutronic transparency.

The key features of the FED-R design are summarized in Table 3-2. The major system and/or components which must be attached to the vacuum vessel are allocated dedicated bays (i.e., spaces between TF coils). This provides a straightforward interface between the tokamak device and the auxiliary systems. The dedicated test module allocations are shown in Fig. 3-4.

The radial build dimensions of the FED-R design are summarized in Fig. 3-5. The dimensions include space allocation for all components as well as for the required gaps for assembly tolerance.

The following sections summarize the main features of the FED-R design.

3.2.1 Magnet System

The magnetic system consists of 12 TF coils, six EF coils and a central OH solenoid. The TF coils are essentially rectangular with rounded corners and are constructed from water-cooled copper plates. The outer leg of each coil is provided with a pair of readily demountable joints to facilitate its removal for maintaining the components placed in the TF coil bore. Gravity support is provided by the trusses located beneath the outer leg of each TF coil. "Window" type openings are provided between adjacent TF coils to access vacuum vessel, test modules, shielding, and poloidal divertor. A common gravity support is used for the inboard TF coil legs and the OH solenoid. TF coil centering forces are reacted by coil wedging. The out-of-plane loads are reacted by the coil case and intercoil support structure.

Table 3-2. Key features of FED-R configuration

Magnetics

- Poloidal field system consists of two superconducting and four normal conducting copper coils.
- Twelve copper TF coils with demountable outer legs.
- Gravity support trusses located beneath the outer leg of each TF coil to support the coil dead weight.
- A "window" type opening is maintained between adjacent TF coils to provide access to vacuum vessel, test modules, shielding, and poloidal divertor modules.
- Common gravity support to the OH solenoid and the inboard TF coil leg.
- TF coil centering forces are reacted by coil wedging.
- OH solenoid is a Bitter plate design.
- TF coil case and intercoil support structure are used to react out-of-plane loads.

Torus

- Torus vacuum vessel is divided into six 60-degree sectors for ease of construction.
  - Twelve, two-part single-null poloidal divertor modules are located in the bottom of the vacuum vessel.
  - Shielding and vacuum vessel rest on a circular platform supported by twelve columns.
  - All bulk shielding is outside vacuum boundary.
  - Vacuum vessel can be easily heated for detritiation at 250°C.
  - Helium cooled outboard vacuum vessel wall for maximum neutron transparency.
  - Water-cooled inboard, top and bottom vacuum vessel walls.
  - 15 locations for major nuclear test modules.
-

COMPONENT	TEST MODULE POSITION	
	STAGE I	STAGE II
HEATING	14,15,18,19,22,23	10,11,14,15,18,19,22,23
FUELING	15,19	15,19
TESTING	4,5,6,7,8,9,10,11,12,13,16,17,20,21,24	4,5,6,7,8,9,12,13,16,17,20,21,24
DIAGNOSTICS, INSTRUMENTATION AND CONTROL	1,2,3	1,2,3

Fig. 3-4. Test module allocation.

*ALL DIMENSIONS IN cm

Fig. 3-5. FED-R radial buildup.

The OH solenoid employs Bitter-plate type winding cooled with water. The EF coils located inside the TF coil bore are normal-conducting and those located outside the bore are superconducting. Each normal EF coil is provided with at least two demountable joints which are needed for installation and replacement (if necessary). The superconducting coils employ 50 kA cable conductor and are cooled with 4.2°K helium. These coils are similar to those employed in the FED baseline.^{1,2}

The elevation views, Figs. 3-1 and 3-3, illustrate the structural design of the TF coil and intercoil structure. The outer TF coil leg is encased with 5-cm of Nitronic 33 to react the out-of-plane loads. In addition, intercoil support structure, in the form of deep box beams, is used to support the coils against overturning moments.

3.2.2 Vacuum Vessel

Figure 3-6 illustrates the vacuum vessel configuration developed for FED-R. The vacuum vessel assembly consists of six vacuum vessel sectors and six internal shield assemblies which protect the seal welds from direct plasma exposure. The vacuum vessel is the first wall. It contains the armor and provides the boundary for the poloidal divertor. The first wall must operate in a high neutron flux environment and is exposed to potential plasma disruptions. The outboard wall is helium cooled for maximum neutron transparency. The inboard, top, and bottom walls are water cooled. To provide for ready access to components, openings are placed in the vacuum vessel. In situ armor tile inspection and replacement is possible. The divertor consists of 12 pairs of modules, with each module pair located between adjacent TF coils. This arrangement allows for removal of these components without disturbing the TF coils. The vacuum vessel is supported by a platform, mounted on 12 columns, located under the torus. Another platform is provided at the same elevation to provide lateral restraint through the TF "window" and is attached to the reactor building floor.

The vacuum pump duct, located under each divertor module, is common with the platform support column and branches to the pump module below the building main floor. A pair of vacuum pump modules, each consisting

Fig. 3-6. FED-R vacuum vessel configuration.

of a turbomolecular pump, scroll pump, and valves, is connected to the vacuum duct with an isolation valve. The peripheral equipment attached to the vacuum vessel is supported by the shielding.

Replacement of a vacuum vessel sector requires prior removal of three outboard TF coil legs, removal of the shield modules in front of the sector, removal of 2 pairs of poloidal divertor modules, and cutting of the vessel sector joints.

3.2.3 Bulk Shielding

The FED-R bulk shielding is located totally outside the vacuum boundary, except for seal weld protection. The shield is designed for minimum thickness. It employs layers of Nitronic 33 and water in varying amounts. The shielding is actively water cooled. Each bulk shield component is treated as a pressure vessel with its independent coolant system.

The bulk shield is designed to be self-supporting. This is accomplished by requiring that the free-standing vertical shield walls support the top shield panels. Also, the bottom bulk shield is used to support the vacuum vessel. To reduce the size of the TF coils, the support platform is used for bulk shielding.

3.2.4 Divertor

A single-null poloidal divertor is chosen for FED-R. The divertor chamber is located at the bottom of the toroidal plasma chamber. This results in an upward displacement of the plasma of 0.5 m relative to the TF coil horizontal center line. The divertor collector is subject to potentially severe particle erosion and, hence, its design includes provisions for rapid replacement. Modular divertor sectors, Fig. 3-2, have been designed which can be removed by radial extraction. The configuration of the divertor modules is indicated in Fig. 3-7. It consists of 12 divertor module pairs, each pair located inside the vacuum vessel and provided with a vacuum pumping system. The divertor module can be removed without dismantling the entire vacuum pumping

Fig. 3-7. Divertor module plan configuration.

system; each divertor module, including its shielding plug, extends outwards to the outer boundary of the torus system where its vacuum seal flange may be readily disconnected. The module extraction motions are indicated in Fig. 3-8. Chamfering the corners of the neutral beam enclosure greatly simplifies the extraction kinematics. The collector plates and other divertor surfaces subjected to high erosion are protected by water-cooled tiles.

3.2.5 Heating

The heating system includes the rf startup assist. This is provided by 2 MW of ECRH (~ 93 GHz) through 24 launchers; two ECRH waveguides are located in each TF "window" (Fig. 3-2). The ECRH heats the plasma to a temperature of over 10 eV (50 eV in the resonance layer). The Oh startup can then be accomplished with a plasma loop voltage of less than 15 volts.

The primary plasma heating method is neutral-beam injectors. For Stage I, three positive-ion based neutral beams provide 50 MW of injected power. The beam injectors are located in three alternate TF-coil "windows" (Fig. 3-2). For Stage II, four negative-ion based neutral beams provide 100 MW of injected power (Fig. 3-3).

3.2.6 Ducts and Ancillary Equipment Shielding

The vacuum pumping duct shield and fueler shielding are similar in construction to the bulk shielding. Layers of Nitronic 33 with water cooling are used. The water serves the dual function of coolant and shielding material.

The portion of the neutral beam duct extending from plasma chamber to isolation valve also has a Nitronic 33/water shield. Outboard of the isolation valve and the beam box the shielding is borated concrete.

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Fig. 3-8. Divertor removal at NB locations.

REFERENCES

1. D. L. Jassby, R. A. Jacobsen, J. Kalnavar.s, L. S. Mason, J. Sekot, *Resistive Demountable Toroidal-Field Coils for Tokamak Reactors*, PPPL-1809, Plasma Physics Laboratory, Princeton Univerisity (1981).
2. J. Kalnavarns and D. L. Jassby, *Resistive Toroidal-Field Coils for Tokamak Reactors*, PPPL-1685, Plasma Physics Laboratory, Princeton University (1980).
3. *Fusion Engineering Device*, DOE/TIC-11600, October (1981).
4. C. A. Flanagan, D. Steiner, and G. E. Smith, *Fusion Engineering Device Design Description*, ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee, December (1981).
5. D. J. Strickler, Y-K. M. Peng, and T. G. Brown, *Equilibrium Field Coil Concepts for INTOR*, ORNL/TM-7746, Oak Ridge National Laboratory, (August 1981).
6. K. W. Ehlers, K. N. Leung, "Characteristics of a Self-Extraction Negative Ion Source," *Proc. Second International Symposium on the Production and Neutralization of Negative Hydrogen Ion and Beams*, Brookhaven National Laboratory, Upton, N.Y., October 6-10, 1980, p. 198.

4. MAGNETIC SYSTEMS

The magnetic systems provide both plasma confinement and control functions. Both the structural design and the overall configuration of FED-R are dominated by the requirements of the magnetic system. These systems constitute a substantial portion (~25%) of the FED-R capital costs. Normal toroidal field (TF) coils were specified as a design requirement. The purpose of this study is to evaluate their suitability for an FED-like machine. Accordingly, TF coils and most of the poloidal field (PF) coils employ a copper conductor cooled with water. Two external PF coils (large radius ring coils) are superconducting, which saves more than 50 MW of power loss. The configuration of TF coils and PF coils is shown in Fig. 4-1. The major design and performance parameters are summarized in Table 4-1.

The FED magnets are designed to provide 4.0 T on axis during Stage I and 5.0 T on axis during Stage II operations, the latter corresponding to 8.8 T peak field at the TF coil windings. The ohmic heating (OH) coil is designed to operate at fields up to ~9.5 T; this magnet employs a water-cooled Bitter-plate winding.

The TF coils are constructed from large size copper plates. Development work will be required to assure that copper plates of the required thickness and size, having satisfactory mechanical properties, can be fabricated.

4.1 TOROIDAL FIELD COIL SYSTEM

The toroidal field coils are designed to be demountable to provide easy access to the internal components of the reactor. The demountability requirement dictates the use of water-cooled resistive TF coils.

The resistive TF coils are constructed from plate segments which can be factory-built and shipped to the reactor site for assembly. The principal disadvantage of resistive TF coils is that they dissipate large power (typically hundreds of megawatts), which results in high operating cost.

COIL	POSITION (center)		MAXIMUM MAT	
	R(m)	Z(m)	STAGE I	STAGE II
EF ₁	2.5	2.5	2.30	2.80
EF ₂	8.0	2.5	-1.92	-2.20
EF ₃	8.0	-3.8	-8.61	-10.79
EF ₄₁	3.5	-3.5	2.55	3.20
EF ₄₂	3.0	-3.5	2.55	3.20
EF ₄₃	2.5	-3.5	2.55	3.20
OH	0.5	-3.0	45.00	45.00

Fig. 4-1. Toroidal-field and poloidal-field coil configuration.

Table 4-1. FED-R magnetic system parameters

Description	Unit	Stage I	Common value	Stage II
Major axis	m		3.5	
Field on axis	T	4.0		5.0
Number of pulses		6000		3300
<u>TF coils</u>				
Number			12	
Conductor			Copper, water-cooled	
Design field at winding	T	7.0		8.8
Winding bore	m		3.7 x 7.2	
Maximum permissible radiation dose	rads		10 ¹¹	
Ampere-turns per coil	MAT	5.8		7.3
Operating current	kA	340		430
Stored energy per coil	MJ	300		470
Coil mean perimeter	m		26	
Resistive loss	MW	175		270
<u>OH coil</u>				
Volt-seconds	Wb		10	
Conductor			Copper, water-cooled	
Maximum field at coil	T	9.5		9.5
Charging time (+9.5 to -9.5)	s	20		20
Length of central solenoid	m		6	
Winding current density	A/cm ²		3750	
Stored energy in the solenoid	MJ		190	
Resistive loss (average during pulse)	MW		1	
<u>EF coils</u>				
Volt-seconds	Wb	10		10
Conductor				
- superconducting coils (2)			NbTi	
- normal coils (6)			Cu	
Field at EF coils	T		<7	
Charging time	s		20	
Discharging time	s		20	
Resistive loss (average during pulse)	MW	20		30

This section describes the TF coil system requirements and design features, the TF coil layout, and the principal components. Alternate concepts are discussed, and the key unresolved issues are identified.

4.1.1 Summary of Requirements and Design Features

The FED-R has twelve toroidal field (TF) coils. Each coil is rectangular with rounded inside corners and has a clear bore height and width of 7.2 m and 3.7 m, respectively. The outer leg of each coil is made removable by incorporation of demountable joints in the horizontal segments of the coil. The number and size of the TF coils are selected on the basis of trade studies involving power consumption, physics requirements, vacuum vessel geometry, and maintainability constraints. The TF coils are designed to operate in a quasi-steady-state mode at a peak field of 7 T during Stage I and 8.8 T during Stage II.

The major requirements and parameters of the TF coils are summarized in Table 4-2. Each coil has 17 turns constructed from flat copper plates. The plates are cooled by water flowing through tubes embedded in each side of the plates. Individual turns (plates) are insulated with the SPINEL-type ceramic $MgAl_2O_4$, which is applied by plasma arc spraying.

Each TF coil is subjected to in-plane forces due to interaction of the coil current with the toroidal field. In addition, out-of-plane loads are caused by interaction of the TF coil current with the radial field component of the poloidal field system. The force containment structure must provide support for the winding. The coil support structure must support the coils against -

- centering forces tending to push each TF coil to the torus centerline,
- out-of-plane forces tending to tip the coils sideways during pulsing action,
- out-of-plane forces tending to bring two adjacent coils together if TF coil currents are unbalanced,

Table 4-2. FED-R TF coil requirements and parameters

• Plasma major radius	3.5 m
• Field on plasma axis	4.0 T (7-T operation) 5.0 T (8.8-T operation)
• TF coils must withstand the following loads:	
• In-plane Lorentz force (steady state)	
• Out-of-plane loads due to PF coil interaction (pulsed)	
-6000 pulses (7-T operation)	
-3300 pulses (8.8-T operation)	
• Out-of-plane loads due to unequal currents in TF coils	
• Dead weight of TF coils plus dead weight of components supported off the TF coils	
• 1 g horizontal and/or vertical seismic load	
• Winding insulation must withstand 10^{11} rads neutron fluence, accumulated over the lifetime of the machine	
• Winding bore size, m	3.7×7.2
• Mean coil perimeter, m	26
• Total I ² R losses, MW	175 at 7 T 270 at 8.8 T
• Weight per coil, tonnes	
Winding	368
Case	33
Intercoil support structure, per coil *	34
Total	435
• Packing fraction	≥ 0.90

* 1/12 total.

- gravity loads, and
- horizontal and/or vertical seismic loads.

Loads are categorized as normal and abnormal. The normal category includes loads occurring in the course of the device performing its intended function. It also includes loads which are known, but which will occur at unanticipated times (e.g., due to plasma disruption). The stress limits are selected to provide reasonable assurance that there will be no equipment damage. Although there may be localized yielding at stress concentrations, there will be no significant yielding. The abnormal category includes loads which may occur during service but which are not expected during normal operation of the device (e.g., a seismic event, loads due to unbalanced TF coil currents). The stress limits for this category permit yielding at structural discontinuities but prevent gross yielding. Following an abnormal event, the device must remain functional.

Structural support members are designed according to structural design criteria which are used for all magnet system components for which the FEDC has design responsibility. These criteria are similar to those used in other large magnet programs such as Princeton TFTR, the Large Coil Program (LCP), the Component Development and Integration Facility (CDIF), and the Mirror Fusion Test Facility (MFTF). The criteria, which are summarized in Table 4-3, provide a safe margin against failure due to gross overloading of a structural member and also provide a margin against growth of an undetected flaw due to cyclic loading and potential subsequent fracture in the vicinity of the flaw.

4.1.2 Coil Layout and Principal Components

This section describes the configuration of the TF coils and their support structure. Several candidate coil shapes were considered, viz., rectangular, D-shape, and rectangular with rounded corners; the selection of a shape considered structural criteria (bending stresses from in-plane Lorentz forces), access between TF coils, and coil perimeter (which affects power dissipation).

Table 4-3. FED structural design criteria used for FED-R

- PRIMARY STRESS LIMITS

Limits are defined as multiples of S_m , defined as follows:

Metals S_m = the lesser of 2/3 yield strength or 1/3 ultimate strength at operating temperature

Nonmetallics S_m = 1/3 ultimate strength at operating temperature

- Normal operating conditions

Primary membrane stress intensity $\leq S_m$

Primary membrane plus bending stress intensity $\leq 1.5 S_m$

Average shear stress $\leq 0.6 S_m$ (metals)

- Abnormal operating conditions

Each of the above limits is multiplied by 1.5

- If buckling is a potential failure mode, a margin of 5 against elastic buckling is required

- FATIGUE AND FRACTURE MECHANICS LIMITS

An allowable peak tensile stress is derived from the Paris crack growth law and from fundamental fracture mechanics principles

For 304 stainless steel (similar to the Nitronic 33 used in FED-R), the combination of 6000 Stage I pulses plus 3300 Stage II pulses results in an allowable peak tensile stress of 69 ksi for Stage II operation and 44 ksi for Stage I operation

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Figure 4-2 shows an elevation view and cross section view of the TF coils. The coils are rectangular window frame shape but with 1 m radius fillets in the inside corners. Each coil contains 17 turns, each fabricated from plate segments 6.1 cm thick for a total thickness of 1.035 m. In the inboard region, the 12 TF coils nest together; the resulting wedging action is the principal means for reacting the net centering force on the coils. To achieve the required fitup of the plates in the inboard region, the plates must be machined to a tapered thickness ranging from 6.1 cm to 2.2 cm at the most inboard point.

There are three types of joints in each of the 17 plates of the TF coils, as shown in Fig. 4-3. These are manufacturing lap joints, electrical joints, and demountable finger joints. Since the overall size of the plates far exceeds the maximum size that can be rolled in the required 6.1 cm thickness, the plates are assembled from subplates, each not exceeding about 1.8 m $\times$ 4.5 m. The subplates are joined using lap joints in which shear pins approximately 2 cm diameter are inserted with a shrink fit.

The electrical joint shown in Fig. 4-3 provides a transition between adjacent turns, which are connected in series. This electrical joint concept is similar to that developed for the ISX-B device.

Figure 4-3 shows the location of the demountable joints in the upper and lower arms of the TF coil. These joints permit removal of the outboard segment of the coil, thereby giving improved access to the components inside the bore of the TF coil. Figure 4-4 shows the details of the demountable joint. The concept is similar to that used in Doublet III. There are 50 pairs of fingers in each joint. In each slot between adjacent fingers, there is an insert bar which bridges the break between the mating pairs of fingers. The insert bars are made of CDA-175 (copper alloyed with beryllium and zirconium), which has a much higher mechanical strength and electrical resistance than the CDA-110 base metal. The high mechanical strength is necessary to transmit the tensile load (due to in-plane Lorentz forces) across the break with a reduced cross section. Each joint in each of the 17 turns is held together by four tension rods (through-bolts) which are made of A286 and

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(ALL DIMENSIONS IN METERS)

Fig. 4-2. TF window frame coils.

Fig. 4-3. TF coil joint locations and coolant schematic.

Fig. 4-4. Demountable finger joint.

preloaded to provide an average contact pressure of about 1500 psi for low electrical resistance.

It is necessary to maintain tight dimensional tolerances during manufacturing of the finger joint. Particularly important is a snug fit of the insert bars in the slots between fingers. If there is a loose fit of the insert bars in the slots, preloading of the tension rods to obtain good electrical contact will lead to unacceptable bending stresses in the fingers.

Figure 4-3 also shows the arrangement of coolant channels in the plates. There are five coolant channel slots milled into each side of each plate in the 17-turn stack; the slots are 1.25 cm x 1.25 cm (1/2 in. x 1/2 in.). The cooling system is sized to accommodate the combined heat load resulting from resistive power dissipation and nuclear heating with a water temperature rise not exceeding 100 F° and a pressure drop not exceeding 100 psi.

The radiation fluence requirement of 10^{11} rads is specified for reducing the major radius of the machine. This is far above the fluence at which organic insulators such as G-10 lose most of their mechanical strength. Ceramic insulators are more suitable for withstanding the high radiation environment. Since the spatial configuration of the coils rules out free-standing or powdered insulation, a sprayed-on coating appears well suited in this instance. SPINEL (MgAl_2O_4) is proposed for FED-R because it is known to suffer very little degradation in mechanical strength under neutron irradiation (see Appendix D). It can be applied to metal surfaces by plasma arc spraying; the metal surfaces can be roughened by blasting with aluminum oxide grit to improve adhesion of the SPINEL. Both contact surfaces will be sprayed with insulation for added protection against electrical shorts in the event of cracking or spalling of the insulation. The sprayed-on insulation thickness is tentatively specified as 0.05 mm (2 mils), which is based on the breakdown strength of the insulator and the FED-R turn-to-turn voltage drop. Testing will be required to determine the optimum thickness for the mechanical loads expected during machine operation. The required development work to qualify the material and application method is discussed in Section 11.

The copper plates in the TF coils are self-supporting against the in-plane Lorentz forces. However, additional support structure is needed to enable the coils to withstand the out-of-plane loads and the resulting overturning moments. A coil case is provided to support the portion of the coil outboard of the demountable joint. In addition, intercoil support structure in the form of deep box-like modules is used to support the coils against overturning moments. A vertical window opening of height 5.4 m is maintained between TF coils to permit replacement of torus sectors.

The coil case and intercoil support structure, shown in Fig. 4-5, are fabricated from 6-cm thick Nitronic 33 steel; this alloy has mechanical properties similar to 304 stainless steel but is better suited to a radiation environment because of its low nickel content.

4.1.3 Assembly of a TF Coil

Each TF coil is composed of a stack of 17 plates and each of these plates is composed of 11 segments (per Fig. 4-3). Because of the large size of the assembled plates and the large number of segments per plate, it is unrealistic to expect the stack of 17 plates to lie flat together during assembly of a coil unless steps are taken to hold them together.

In the outboard region, the plates are surrounded by a welded case, while in the upper and lower legs the yokes hold the stack together. However, in the inboard region where the coils wedge together, there is no room for a case. In this region, adjacent plates in the stack are bonded together using an epoxy resin. This bonding operation is performed prior to moving the TF coil into final position, so that external C-clamps may be used to hold the stack together until the resin cures. A similar bonding technique was used in the assembly of the Doublet III B-coils.

It is true that the mechanical strength of the epoxy will be gradually degraded by high radiation in the inboard region. However, this is of no consequence, since the epoxy will have already done its job. Once the TF coils are in their final position in the tokamak array and are then energized, the net centering force on the coils holds the plates in place by wedging action.

Fig. 4-5. Coil case and intercoil support structure.

4.1.4 Alternate Concepts

This section describes the main alternate concepts which were considered in arriving at the present FED-R design.

TF coil shape

At first, a rectangular window frame shape with sharp inside corners was considered. As expected, the bending stresses in the corners were unacceptable. A D-shaped TF coil was then considered. This shape is approximately a constant-tension shape and has very low in-plane bending stresses. Furthermore, the coil perimeter is about 6 percent less than that of the window frame coil, which reduces power consumption by the same amount. However, the D-shape affords poorer access between TF coils. A compromise shape which has manageable in-plane bending stresses in the corners without impeding access between coils is a rectangular window frame but with generously rounded inside corners (1 m radius). This shape was selected for the present design; calculation of in-plane stresses using frame theory demonstrated acceptability of this shape.

Demountable joint concept

As an alternate to the demountable finger joint design discussed earlier, a demountable lap joint was considered. Two versions of the lap joint were considered: (1) a version using through-bolts passing through the lap joints in all 17 turns, (2) a version using a separate set of bolts in individual turns. The version using through-bolts was rejected because of the difficulty of inserting long bolts through the stack of plates from the side, given the limited amount of space between TF coils. The version using separate bolts in individual turns was rejected because the large number of bolts would make assembly or disassembly of the joint very tedious.

4.1.5 Open Issues Relating to TF Coils

This section discusses some unresolved technical issues relating to design and fabrication of the TF coils.

Insulation material

The feasibility of sprayed-on SPINEL insulation must be confirmed by testing. While the ability to withstand high radiation fluence is not in serious doubt, there is uncertainty whether a sprayed-on insulating material several mils thick could withstand the mechanical strains which are developed in the copper plates. Furthermore, in view of the several thousand pulsed applications of the out-of-plane loads, the ability of the insulator to resist wear must be demonstrated.

Machining tolerances

Fabrication of the TF coil plate segments which make up the inboard region of the coils requires bending 6 cm thick plate material to accommodate the cross section transition from rectangular to trapezoidal. The fabrication also requires machining these plates to a tapered thickness. This machining must be done to tight dimensional tolerances in order to obtain good fitup of the plates in the inboard region. The ability to hold such tolerances during manufacture must be demonstrated.

Demountable joint

The complexity of the demountable joint design, and the time required for assembly or disassembly, is a major feasibility issue. It has been estimated that several hundred hours would be required to disassemble and reassemble the present demountable joint (see Sect. 9.2). While this time may be considered undesirable, an alternate concept which would permit faster disassembly or reassembly has not been identified.

While the FED-R demountable joint design resembles the joint which has been built in Doublet III, the Doublet III joint has not actually been demounted after the coils were energized. However, all the

Doublet III coils will be disassembled in 1984 in order to install a new vacuum vessel, and this action will, to some extent, constitute a demonstration of the feasibility of the FED-R joint design.

4.2 POLOIDAL FIELD COIL SYSTEM

The poloidal field (PF) coils include several coil sets. Together, these coils provide the functions of plasma initiation, heating, shaping, and equilibrium. From a configuration standpoint they are grouped as a solenoid (long compared to diameter) and ring coils (diameter large compared to coil cross section).

Equilibrium field (EF) coils shape the plasma in cross section and provide the vertical field necessary to maintain it in equilibrium. The function of ohmic heating (OH) coils in FED-R is to inductively start up the plasma current and provide modest heating near the beginning of the cycle. Small control coils are used in conjunction with feedback control circuits to control the position of the field null for plasma initiation and to control the plasma position radially and vertically during the cycle. These coils have not yet been designed and do not appear in the drawings.

The PF coil locations and current ratings are shown in Fig. 4-1. This section includes discussions on PF configuration development and design of individual coils.

4.2.1 Poloidal Field Configuration

The FED-R poloidal field (PF) coil system has been selected to satisfy torus space and maintenance constraints; the MHD equilibrium requirements implied by impurity control (expanding boundary/poloidal divertor); and plasma position/shape control. The system consists of a central solenoid, copper shaping coil groups internal to the TF coils, and outer superconducting equilibrium field (EF) coils external to the TF coils. The shaping coils maintain a D-shaped plasma with a single null point defining the plasma boundary, as shown in Fig. 4-6. The outboard EF coils provide the main vertical field necessary for MHD

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Fig. 4-6. Equilibrium PF contours (a) low β , $\langle \beta \rangle = 1\%$, (b) high β , $\langle \beta \rangle = 6\%$.

equilibrium and are made superconducting to eliminate large power losses associated with normal coils.

EF coil currents are obtained by solving a free-boundary tokamak equilibrium problem while iteratively adjusting the currents in some subset of the given coil locations to center the plasma.¹ In the case of FED-R, currents in the central solenoid (OH) and shaping coils [EF1, EF4 (1-3)] are fixed in each calculation, based on the operation scenario (i.e., a 20-s OH current swing from 30 MA to -30 MA, then to 0 MA at 100 s), and the requirement of a poloidal separatrix, while outer EF coil currents (EF2, EF3) are varied during the calculation to position the plasma. (Note: The peak OH current was later increased to 45 MA.)

The elongation and triangularity of the plasma boundary above the plasma midplane are approximately $K = 1.5$ and $\delta = 0.2$, limited somewhat by the requirement of a connected scrape-off.^{2,3} The active null point below the plasma midplane is positioned at $R_s = 3.2$ m and $Z_s = -1.6$ m and varies radially by a maximum of 5 cm and vertically by a maximum of 4 cm from this location during operation. The time dependence of PF coil currents and the plasma current are shown in Fig. 4-7 (Stage I) and Fig. 4-8 (Stage II).

4.2.2 Solenoid Layout

The ohmic heating (OH) solenoid is required to provide a flux linkage of 10 volt-seconds when pulsed from +I to -I. The OH solenoid configuration is shown in Fig. 4-9, along with major engineering parameters.

The inside and outside radii of the solenoid winding are 40 cm and 60 cm, respectively, and the axial length of the winding is 6 m. A 10-cm thick stainless steel sleeve is applied over the winding to react the radial bursting load. The peak conductor current density is 4700 A/cm². Peak ampere-turn capability of the solenoid is 45 MA (studies of Sect. 4.2.1 assume 30 MA solenoid). The OH solenoid will deliver 11 volt-seconds at the plasma with full bi-polar swing of the current. The OH solenoid current is programmed to zero during the first 100 s of the

Fig. 4-7. PF coil and plasma currents as a function of time for Stage I.

Fig. 4-8. PF coil and plasma currents as a function of time for Stage II.

- AMPERE TURNS 45 MA
- VOLT-SECONDS 10 V-s
- PEAK FIELD 9.5 T
- TIME CONSTANT OF CASING 0.045 s
- COOLANT TEMPERATURE RISE 100 °F
- COOLANT FLOW RATE 5140 gpm

ALL DIMENSIONS IN cm

Fig. 4-9. OH solenoid concept utilizing Bitter plates.

long FED-R pulses, as shown in Figs. 4-7 and 4-8. Neutral beam current drive compensates for the back emf caused by this process.

The solenoid winding employs Bitter plates. The winding is cooled with water flowing through 420 axial holes (each hole is 1.25 cm in diameter). The water flow velocity is 6.5 m/s, the water pressure drop is ~ 100 psig, and the water temperature rise is $\sim 60^\circ\text{C}$. The time constant of the 10-cm stainless steel sleeve applied over the winding is 0.07 s, which is small compared with the 20-s plasma startup time and is, therefore, not critical from the viewpoint of causing delay during the startup. The energy dissipated during the startup in the sleeve is less than 1 percent of the peak energy stored in the OH solenoid.

4.2.3 Exterior (Superconducting) Ring Coil Layout

The exterior ring coils will be superconducting to eliminate large ohmic power consumption. These coils are located in a region of low nuclear radiation intensity, so that adequate shielding is straightforward. The major design parameters for these coils are summarized in Table 4-4. A conceptual drawing for the ring coils is shown in Fig. 4-10.

Key design features of the superconducting ring coils are

- 50 kA cable conductor;
- pancake winding;
- stainless steel, ring stiffened case to provide structural support and contain helium;
- pedestal supports carry dead weight, seismic loads and out-of-plane magnetic loads; coil case is designed to equilibrate magnetic bursting load and magnetic bending loads; and
- dielectric breaks in the case to reduce induced poloidal currents; joints sealed by welded omega seals or double "O" rings.

The winding and case design is identical to that of the superconducting ring coils employed in the FED baseline.³ However, the FED-R superconducting ring coils are supported by columns in the plane of the TF coils, which run directly to the floor.

Table 4-4. Superconducting ring coil design parameter

Parameter	Coil EF ₂ ^a	Coil EF ₃ ^a
<u>General</u>		
Mean radius, m	8.00	8.00
Radial build, m		
Winding	0.4	1.0
Overall	0.8	1.6
Axial build, m		
Winding	0.4	1.0
Overall	0.8	1.6
Weight, 10 ³ kg		
Winding	23	110
Structure	9	45
<u>Electromagnetic</u>		
Maximum field at winding, T	~3	~5
Conductor rating, kA	50	50
Winding type	Pancake	Pancake
Total number of turns	40	196
Current density, A/cm ²		
Winding	1250	980
Overall	310	380
<u>Performance</u>		
Charge time, s	20	20
Stored energy, MJ	83	1700

^aSee Fig. 4-1 for coil locations.

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Fig. 4-10. Superconducting ring coil concept.

4.2.4 Internal (Normal) Ring Coils

The internal normal ring coil design reflects the requirement for demountable mechanical joints to facilitate initial assembly and coil replacement (if necessary). The four internal coils (EF1, EF41, EF42, and EF43, as shown in Fig. 4-1) are all normal. The major design parameters for these coils are summarized in Table 4-5.

Each normal coil turn will have at least two demountable joints. One possible joint concept is shown in Fig. 4-11. Design of these coils is similar to the normal coils employed in the FED-Baseline design.³ No component designs were developed for this study.

Table 4-5. Normal ring coil design parameters

	Coil EF ₁ ^a	Coils EF ₄₁ , EF ₄₂ , EF ₄₃
Mean radius, m	2.5	3.0 (EF ₄₂)
Radial build, m	0.4	0.4
Axial build, m	0.4	0.4
Total current, MAT ^b	2.2	2.5
Number of turns	22	25
Power consumption, MW ^b	10	14 (each coil)
Conductor current density, A/cm ^{2b}	1400	1500
Weight, 10 ³ kg	20	74 (all)

^a See Fig. 4-1 for coil locations.

^b Stage I.

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- EMPLOYED IN CURRENT FED BASELINE CONCEPT
- LOW RESISTANCE RUGGED JOINT
- TEDIOUS FOR ASSEMBLY AND DISASSEMBLY
- IMPROVEMENTS DESIRABLE TO REDUCE REPLACEMENT TIME
- THROUGH BOLTS CAN BE CONSIDERED IN PLACE OF BOLTS FOR INDIVIDUAL TURNS

Fig. 4-11. Individual turn bolted joint concept.

REFERENCES

1. D. L. Jassby, R. A. Jacobsen, J. Kalnavarns, L. S. Masson, J. Sekot, *Resistive Demountable Toroidal-Field Coils for Tokamak Reactors*, PPPL-1809, Plasma Physics Laboratory, Princeton University (1981).
2. J. Kalnavarns and D. L. Jassby, *Resistive Toroidal-Field Coils for Tokamak Reactors*, PPPL-1685, Plasma Physics Laboratory, Princeton University (1980).
3. *Fusion Engineering Device*, DOE/TIC-11600, October 1981.
4. C. A. Flanagan, D. Steiner, and G. E. Smith, *Fusion Engineering Device Design Description*, ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee, December 1981.
5. D. J. Strickler, Y-K. M. Peng, and T. G. Brown, *Equilibrium Field Coil Concepts for INTOR*, ORNL/TM-7746, Oak Ridge National Laboratory, August 1981.
6. L. G. Davis and J. M. Rawls, "Modification of Doublet III," in *Proc. 9th Symp. Engineering Problems of Fusion Research* (IEEE, Chicago, 1981), pp. 1939-42.

5. ELECTRICAL SYSTEMS DESIGN

The electrical system consists of the plasma heating equipment, the power convertors, and the information and control (I&C) system. The heating system includes the rf startup assist. This is provided by 2 MW of ECRH (~ 93 GHz) through 24 launchers. Because the ECRH heats the plasma to a temperature of over 10 eV, and on some flux surfaces to over 50 eV, startup can be obtained with a plasma loop voltage of less than 15 volts. The startup coils and feedback control coils are assumed to be of negligible size and cost.

The electrical power is provided by the utility grid. Approximately 500 MVA of pulsed power is supplied directly from a utility line and 600 MVA of additional pulsed power is provided by the energy storage motor generator flywheel sets (MGF). The MGF units require 30,000 HP drive capacity with 600 MVA peak rating and 6.4 GJ stored energy. The TF coils operate in steady state while the PF system is pulsed during each plasma burn.

The facility is operated using the latest digital control systems and data handling procedures.

5.1 PLASMA HEATING SYSTEMS

The auxiliary heating systems which perform the functions of start-up assist, bulk heating, and current drive maintenance are described here. The objective is to summarize the requirements for each function and show how they are satisfied by the design configurations.

RF assisted start up has been studied for the baseline FED^{1,2} in FY81 and in earlier efforts. In this year, a trade-off study between rf power, PF coil volt-seconds, and start-up time was initiated. Longer start-up times were found to be compatible with steady-state or quasi-steady-state operation. For FED-R, a combination of 2 MW ECRH, 10 volt-seconds from the OH, a comparable flux swing from the EF system, and the NB injection bulk heating allow start up in about 20 seconds. For Stage II (with its higher operating toroidal field), the device is started up at the lower field. The field is subsequently raised (over a

few minutes) to the Stage II level. This procedure eliminates the need for a second ECRH frequency.

Neutral beam injection was chosen for the bulk heating and current maintenance functions for the reasons discussed in Sect. 2.2.2. There are two Stages to the program, the first having an operational date of 1990 and the second that of 1998. In Stage I the NBI approach utilizes a positive-ion-based system which offers the advantage of low to moderate development risk. This NBI system is very similar to the one proposed for ETF, FED,^{1,2} and INTOR. For Stage II, the positive-ion-based system is replaced by a high energy negative-ion-based system. The extra eight years (between Stages) permit this advanced beam technology to mature for inclusion in the FED-R program.

The order of presentation is to start with a summary of requirements, then briefly describe the ECRH system which is similar to the FED-Baseline system, and then provide a more detailed discussion of the negative-ion-based NBI.

5.1.1 Plasma Initiation and Start-up

Requirements

The plasma current, density, and temperature in FED-R are to be increased by a series of processes lasting approximately 20 s. During the current build-up phase, the vacuum vessel must serve as an electromagnetic path of fairly long time constant in order to assist the maintenance of a grossly stable and well-centered plasma column. This requirement, among others, has resulted in the specification of 0.1 m Ω for the vessel loop resistance.

The following are the major steps that must occur during start-up. The time variation of important plasma parameters is indicated in Fig. 5-1.

0 to 0.2 s. The filling gas in the torus is ionized in a region of 0.5 m diameter by a pulse of electron cyclotron radiation (ECH) at 93 GHz. This region is centered where $B \approx 3.4$ T, or somewhat outboard of the final magnetic axis of the plasma. If necessary in Stage II, the

ORNL-DWG 82-2281A FED

Fig. S-1. Plasma initiation and start-up.

magnetic field would be lowered at the beginning of each pulse to allow resonance in a convenient region. At the termination of this phase, the plasma parameters will be $T_e > 10$ eV, $n_e \sim 5 \times 10^{12} \text{ cm}^{-3}$, and loop voltage ~ 10 V.

0.2 to 2 s. The OH transformer is activated to establish a 300 kA current, while the plasma radius is increased to its final value of 85 cm, $\bar{n}_e$ is raised to $2 \times 10^{13} \text{ cm}^{-3}$ by gas puffing, and T_e rises to several hundred volts.

2 to 20 s. The neutral-beam injectors are activated and the power is gradually increased to 50 MW in Stage I or to 100 MW in Stage II. At the same time the vertical field is increased to maintain the plasma position as the plasma pressure and current increase. The plasma current is driven to its final value of 3.5 MA in Stage I or 4.4 MA in Stage II by the neutral-beam-induced current as well as by part of the equilibrium field (EF) flux swing. The final current is driven entirely by the neutral beams.

ECRH design description

A waveguide, horn-type launcher is shown in Fig. 5-2 positioned on the high field side of the FED-R vacuum vessel. There are two of these launchers in each sector (24 in all), each carrying ~ 100 kW. With 2 MW required for operation, there are four spare launchers.

Figure 5-2 shows the system component layout taken from the FY81 FED² design. Table 5-1 summarizes the ECRH system. The system features mode control, double tritium containment, and a low loss corrugated waveguide system operating with the hybrid HE_{11} mode. The diagram shows conversion to the circular transverse electric TE_{01}^* mode for the launcher. This is the baseline option. A stepped reflector and horn launcher are required. A second approach would be to continue with the HE_{11} mode in the launcher. In this case, there would be no need for a mode converter nor a stepped reflector. The tube (source) mode is left open since the tube has not been developed for this application.

Each launcher is supplied by a string of components such as this shown in Fig. 5-3. It is expected that the tubes (gyrotrons) will

Fig. 5-2. ECRH RF assist launchers moved on high field side.

Table 5-1. Summary of ECRH characteristics

	Units	Stage I
A. System		
1. Power		
-at launcher	(MW)	2.0
-generated	(MW)	4.8
2. Efficiency	(%)	23
3. Frequency	(GHz)	~93
4. Launcher position		High field side
5. Launched mode		Extraordinary
6. Pulse length	(s)	Steady state
B. RF generator		
1. Power/gyrotron	(MW)	>0.2
2. Number of gyrotron		
-Operational		20
-Reserve		4
3. Efficiency	(%)	>50
C. Transmission system		
1. Type		Corrugated waveguide
2. Mode		HE ₁₁
3. Window		Cooled BeO
4. Launcher		24-mode transducing
5. Efficiency	(%)	>50

Fig. 5-3. ECRH system configuration.

produce about 200 to 250 kW each. After sustaining a ~50 percent loss in the transmission system, the power at each launcher is ~100 kW. The gyrotron oscillator is about 50 percent efficient. Adding a few percent for power supply losses, the overall efficiency should be better than 23 percent.

The gyrotron frequency is fixed (not tunable) by design. If it is desired to operate FED-R at different toroidal field levels or start up the plasma at different positions, there must be a second tube connection.

5.1.2 Bulk Heating and Current Drive Requirements

Attaining the FED-R wall loading objectives, 0.4 MW/m^2 (peak) in Stage I and 1.3 MW/m^2 (peak) in Stage II, leads to the requirement for 50 MW at 150 keV D^0 positive-ion-based steady-state neutral beam in Stage I, to be replaced by 100 MW at 250 keV negative-ion based steady-state neutral beams in Stage II. The NBI requirements are summarized in Table 5-2.

In addition to heating, the neutral beam system must simultaneously drive approximately 4 MA of plasma current. The effect of beam energy, injection angle, and beam species mix on the driven current for FED-R-like plasmas is indicated in Table 5-3, Figs. 5-4 and 5-5, respectively. For Stage I, the 50 MW of 150 keV positive-ion-based power, a large injection angle (45°), a high species mix (90 percent full energy), and optimism with respect to the attainable electron temperature [reference $T_e(0) = 22 \text{ keV}$] are all required for the simultaneous achievement of 0.4 MW/m^2 and 4 MA. Tritium beams are required to fuel the central region. For the same penetration as the D^0 beams, the T^0 beams require a higher energy ($3/2 \times 150 = 225 \text{ keV}$). For Stage II, 100 MW at 250 keV, 4 MA of current drive is easily attained. The "performance margin" for this case will be used to relax the electron temperature requirement and/or to allow use of tritium beams at 300 keV with near normal injection, rather than 375 keV at 45° injection.

The increase in wall loading, in going to Stage II, from 0.4 MW/m^2 to $\sim 1.3 \text{ MW/m}^2$, means both Q and the injected power must be increased. To obtain higher Q requires increases in plasma density which reduces

Table 5-2. NBI bulk heating and current drive maintenance requirements

	Units	Stage I	Stage II
Ion source		D ⁺ , T ⁺	D ⁻ , T ⁻
Energy			
-Deuterium	(keV)	150	250
-Tritium	(keV)	225	300
Power	(MW)	50	100
Injection angle			
-Deuterium	(DEG)	45	45
-Tritium	(DEG)	45	36
Pulse length	(s)	Steady state	Steady state
Number of injectors			
-Deuterium		2	2
-Tritium		1	2

Table 5-3. Effect of beam energy at constant
 I_{bd} and P_w^* in Stage I

E_b	keV	150 (+ fractions)	200	300
P_b	MW	62.5	47	36.5
$n_e(0)$	$10^{13}/\text{cm}^3$	7.0	7.4	8.0
$n_e(0) - E$	$10^{13} \text{ s}/\text{cm}^3$	2.1	2.9	4.1
$\langle \beta_i + \beta_e \rangle$	%	2.8	3.0	3.2
$\langle \beta + \beta_b \rangle$	%	1.7	1.8	2.0
$\langle \beta_{TOT} \rangle$	%	4.5	4.8	5.2
Q		1.3	1.7	2.2

* Demand $I_{bd} = 4 \text{ MA}$ and peak $P_w = 0.4 \text{ MW/m}^2$.

Beam power and plasma density are dependent variables.

$T_e(0) = 22 \text{ keV}$, $T_i(0) = 43 \text{ keV}$, $R_{TANG} = 2.7 \text{ m}$ (40°).

β values refer to B_ϕ ($R = R_C$) = 4.0 T.

$I_{bd} \equiv$ beam driven current.

$P_w \equiv$ neutron wall loading.

ORNL-DWG82-3023 FED

$E_b = 150$ keV $n_e(0) = 7.5 \times 10^{13}$ FULL ENERGY
 $T_i(0) = 43$ keV $T_e(0) = 22$ keV   ION FRACTION = 0.8
 $I_{bd} = 4$ MA $P_w = 0.5$ MW/m²

Fig. 5-4. Effect of injection angle.

ORNL-DWG82-3024 FED

$E_b = 150$ keV $n_e(0) = 7 \times 10^{13}$ $R_{TANG} = 2.7$ m
 $T_i(0) = 43$ keV $T_e(0) = 28$ keV $\theta_{inj} = 40^\circ$
 $I_{bd} = 4$ MA $P_w = 0.5$ MW/m²

Fig. 5-5. Effect of species mix.

the penetration at a given beam energy. The 150 keV $-D^0$ energy level is not sufficient to penetrate the plasma for the Phase II operation. A higher beam energy is required and is also preferable for increasing the current drive efficiency. At the present time, developers feel a 250 keV beam energy is a reasonable next step. Table 5-4 shows performance for the higher beam energies (200-800 keV range). The required electron temperature T_e decreases with an increase in beam energy and the risk of attaining a given T_e is smaller, the lower the required T_e . Finally, the data indicate there is a minimum n_T at approximately 250 keV.

The FED-R geometry places requirements/constraints on the interface design of the NBI. The injection angle is set for tangential injection. This 45° injection angle, coupled with the TF coil geometry, results in a long duct, 6.0 m, between the vacuum vessel and beam pumping box and a rather restrictive 0.8 m duct width. The duct height has been set at 1.05 m because larger values do not appreciably improve beam system performance in the modeling studies. The number of beam injectors is held to three in Stage I and four in Stage II in order to minimize cost and to maximize the wall area available for blanket test modules.

Since there is excess current drive capability in Stage II, the tritium beam can be operated at a more perpendicular injection angle to improve its penetration compared to that of the deuterium.

5.1.3 Neutral Beam Injector Design

For Stage I, the 150-keV positive-ion-based neutral beam injector design configuration is that of the FED baseline alternate. Table 5-5 summarizes the characteristics of this design. Further design description detail can be found in Ref. [2], Volume II. One subject not covered in the reference documentation is the use of tritium in place of deuterium. It is expected that a 225 keV version of the design can be built with comparable performance in terms of efficiencies and power. The cryopump design in the NB injectors will have to be capable of regeneration during operation to limit the tritium inventory.

A top and side view of an FED-R, Stage II neutral beam injector is shown in Figs. 5-6 and 5-7, and characteristics are summarized in

Table 5-4. Effect of beam energy in Stage II

E_b	keV	200	300	400	800
$n_e(0)$	10^{14} cm^{-3}	1.5	1.7	1.8	2.1
$T_e(0)$	keV	22	17	16	14.5
$T_i(0)$	keV	44	34	32	29
τ_E	s	0.29	0.25	0.25	0.27
$\langle \beta_e + \beta_i \rangle$	%	5.9	5.4	5.4	5.6
$\langle \beta + \beta_a \rangle$	%	1.5	1.4	1.4	1.5
$\langle \beta_{\text{tct}} \rangle$	%	5.4	4.8	4.8	5.1
P_{th}	MW	180	180	190	230
Q_{bt}		0.76	0.66	0.58	0.35
P_f	MW	256	246	248	265
$n_e(0)\tau_E$	$10^{14} \text{ cm}^{-3}\text{s}$	0.435	0.425	0.450	0.567

Fixed parameters: $I_{\text{bd}} = 4.4 \text{ MA}$ and $P_w = 1.3\text{-}1.4 \text{ MW/m}^2$

$$R_o = 3.5 \text{ m} \quad n_e(r) = n_e(0) (1 - (r/a)^2)$$

$$a = 0.85 \text{ m} \quad T_e(r) = T_e(0) (1 - (r/a)^2)^2$$

$$b/a = 1.5 \quad T_i(r) = T_i(0) (1 - (r/a)^2)^2$$

$$B_z(R = R_o) = 5.0 \text{ T}$$

$$n_D/n_T = 1 \quad P_z = 100 \text{ MW}$$

$$\Delta z_{\text{eff}}(\text{Fe}) = 0.96 \quad E_1:E_2:E_3 = 1:0:0$$

$$n_{\text{He}}/n_e = 0.02 \quad R_{\text{tang}} = 3.1 \text{ m}$$

$$Z_{\text{eff}} = 2.0 \quad \dots (R_o + a)$$

$$q(a) = \frac{5B_o a (1 + K^2)}{I_p R_o^2} = 2.25$$

Table 5-5. Positive ion NBI characteristics

Source design	
A. Number (to meet 50 MW)	17
B. Size, H x W (cm ²)	16 x 31
C. Current (amp)	67
D. Per injector	6
E. Configuration	2 stacks of 3
F. Divergence (deg)	<0.8
G. Energy (keV)	150
H. Pulse length (s)	Steady state
Interface design	
A. Number of NBI	3
B. Drift duct	
(1) height (m)	1.05
(2) width (m)	0.8
(3) length (cm)	560
(4) valves	2 vacuum
C. Power density average (kW/cm ⁻²)	2.1
D. Injection	
(1) angle (deg)	42 and 48
(2) power per NBI (MW)	18
(3) beam cross over	Drive duct center
E. Efficiency (overall NBI)	
(1) full energy	32
(2) all energies	40
Cryopumps	
A. Gas type	H or H isotopes
B. Area	
(1) drift region (m ²)	32
(2) gas cell region (m ²)	41
C. Configuration	Differential LHe shielded by LN ₂ chevrons
D. Refrigeration load (kW)	≤2
Ion deposition	
A. Type	Direct recovery on full energy
B. Capabilitiy	≥0.6 efficiency
Species mix	
A. Ion fraction	0.9, 0.05, 0.05
B. Power fraction	0.80, 0.09, 0.11

Fig. 5-6. Top view of negative ion NBI for Stage II.

Fig. 5-7. Side view of negative ion NBI for Stage II.

Table 5-6. The negative ions originate at the ion source and are transported through curved openings in the neutron shield to be stripped of their extra electrons by photodetachment in the laser neutralizer. The small fraction of ions, which are not neutralized (~ 3 percent), are swept out of the beam path to be collected at a suitable ion dump, while the neutral beam continues through the transport duct into the reactor plasma.

The beams originate as negative ions emitted from a self-extracting surface conversion negative ion source.³ Because the ions are self-extracting, the beam is relatively free of stray electrons with little power loss. The ions can be brought up to any reasonable energy by means of a conventional electrostatic accelerator.

A 250 keV ribbon beam of deuterium (2 cm wide by 250 cm high), delivering 17.5 A of D⁻ at a current density of 35 mA cm^{-2} is obtained from a single ion source; six such ribbon beams, operating in parallel, form the basis of a neutral beam injector. Similarly, a 250 keV ribbon beam of tritium (2 cm wide by 300 cm long) delivering 17.5 A of T at a current density of 28.5 mA/cm^2 is available from a tritium ion source. Six such ribbon beams are used in a neutral tritium injector.

A series of TFF (transverse-field focusing) components,⁴ similar to that shown in Fig. 5-8, transports the ribbon beams from their associated accelerators, through the neutron shielding, to the neutralizer, via a sequence of curved paths. The transport properties of the TFF components result from focusing obtained when such a beam is subjected to electrostatic deflection.⁵ Adequate magnetic shielding is provided for the ion sources, the TFF components, and in the vicinity of the neutralizers.

The overall configuration shown in Figs. 5-6 and 5-7 features removal of the ion source from direct exposure to the neutron flux originating in the reactor and using the final TFF component to direct each ribbon beam, through the neutralizer, to a common focus in the reactor plasma.

About 97 percent of the negative ions in each of the ribbon beams lose their extra electrons by photo neutralization⁶ in an optimized

Table 5-6. Negative ion NB injector characteristics

	Deuterium	Tritium
<u>Ion source</u>		
Current	17.5 A	17.5 A
Gas efficiency	15%	15%
Fraction of electrons	0.5 %	0.5%
<u>Accelerator</u>		
Current density	35 mA/cm ⁻²	28.5 mA cm ⁻²
Aperature	(2 × 250 cm ²)	(2 × 300 cm ²)
Efficiency	78%	78%
Beam energy	250 keV	250 keV
<u>TFF</u>		
E, transverse field	~20 kV cm ⁻¹	~20 kV cm ⁻¹
R, radius	~25 cm	~25 cm
<u>Laser neutralizer</u>		
Laser efficiency	16%	16%
Input power	656 kW	481 kW
Wavelength	1.3 μm	1.3 μm
Stripping fraction	97%	97%
Neutralizer length	1.8 m	1.5 m
Individual laser power	2.1 kW	1.75 kW
Number of lasers	50	44
Mirror reflectivity	99.9%	99.9%
Mirror dissipation	40 W cm ⁻²	40 W cm ⁻²
Window transparency	99.9%	99.9%
<u>Injector performance</u>		
Current	100 A	100 A
Voltage	250 kV	250 kV
Electrical efficiency	69%	69%

BEAM TRANSPORT VIA
TRANSVERSE FIELD FOCUSING

Fig. 5-8. Transverse field focusing components.

laser neutralizer cell similar to that shown in Fig. 5-9. To accomplish this, each 17.5 A deuterium ribbon beam requires 50 atomic iodine chemical lasers, each dissipating about 2.1 kW in the optical resonator, while each 17.5 A tritium ribbon beam requires only 44 lasers with a dissipation of 1.75 kW. The ions pass through an expanded portion of the laser resonator which is evacuated to a pressure of about 10^{-5} torr, while the laser gain medium, consisting of excited iodine atoms, is confined by a laser window on one end and a back mirror on the other. The atomic iodine chemical laser block diagram is shown in Fig. 5-10.

5.1.4 Alternate Concepts

Source/accelerator

A number of advantages may accrue from the application of a source with a higher current density output. Sources are under development (ORNL and BNL) which operate at an order of magnitude increase in current density (~ 100 mA/cm²), and the source pressure is low enough to permit direct coupled accelerating electrode structures for d.c. operation. Increased output current density will result in an order of magnitude decrease in the required source extraction area and consequently the associated plasma generator cost and complexity. These high current density sources employ extraction transverse to the source magnetic field which inherently produces a beam deflection of 10° to 20° thus avoiding direct line of sight to the neutron flux traversing the entrance duct. In addition, a conventional electrostatic accelerating system is employed. A method to drain off the electrons present in the accelerator gaps, from whatever cause, has already been developed. None of these electrons receive an energy greater than a single gap potential difference. Additional development of the cesium flow control and the convertor cooling mechanism is required to increase pulse length from the 10 s level to d.c.

Neutralizer

The conversion efficiency of $>95\%$ for the photodetachment method proposed makes this neutralization scheme a definite favorite. However,

Fig. 5-9. Laser neutralizer.

Fig. 5-10. Neutral-beam injector using atomic iodine chemical laser for neutralizing D^- .

there are several areas of development required. Some of these are handling of the chemicals of the laser medium; lifetime of the transmission window between the lasing medium and the neutralizing cavity; development of a 99.9% reflective coating which can survive a hostile neutron flux environment; and development of high mode cavity to ensure a uniform photon flux for maximum negative ion neutralization. Two other schemes are available with progressively lower neutralization efficiency: highly ionized plasma with a neutralization efficiency up to about 85% and a gas stripping cell with efficiency of about 60%.

For a plasma to be an effective neutralizer, the plasma must be $\geq 50\%$ ionized. Typical ionization percentages in ion sources lie in the range of 1% to 10%. However, development at both BNL and Novosibirsk appears very promising. Test systems at BNL have operated within 60% to 70% of the target thickness required. Pumping requirements are increased with this system.

The simplest neutralizer is a gas stripping cell. An alkali vapor cell is also a candidate. The advantage of reduced gas handling in the latter is offset by the complexities of vapor handling and contamination. It is not possible to couple a gas cell directly to the ion source/accelerator due to the resultant high neutral density in the accelerator column producing premature stripping of the negative ions. Such stripping constitutes not only a loss in beam current but also an unacceptable power loading on the electrodes and energy straggling in the neutral beam. Consequently, a decoupled gas cell is required and a significant increase in gas handling and potentially an increased reionization loss occur in the drift duct due to neutral gas streaming. Either of the alternate neutralization schemes will operate with the proposed source or the alternate source concepts.

Ion beam removal

With photo-detachment neutralization, power in the unneutralized beam could vary from as low as 1% or 2% to as high as $\sim 10\%$. In addition, there may be $\sim 5\%$ to 10% ionization (positive ions) of the neutral beam on the background gas ($\sim 10^{-5}$ torr) along the path from the vicinity of the neutralizer to the exit of the electrostatic ion beam deflector.

This $\leq 10\%$ of the beam power is dissipated as ions before entering the drift duct. Assuming the unneutralized beam density is 1 mA/cm^2 (5% of the neutral beam density), the plasma density created by the beam has a Debye shielding distance of about 1 cm. Thus, the axial extent of the electrostatic plates will have to be sufficiently long to give field penetration over the $\sim 5 \text{ cm}$ beam width. With this scheme, the plates also serve as a beam dump (with unwanted additional power loss) for part of the beam.

An alternate beam deflector is a magnetic field. The magnetic field could be impressed by a special magnetic circuit or the fringe field of the experimental device since the latter operates steady state. However, both the magnitude and direction of the fringe field must be compatible with beam system constraints for dissipating the ion energy. Otherwise, a special magnetic circuit must be employed. Magnetic shielding of the fringe field in this approach is necessary to adequately control termination of both the positive and negative ion components of the unneutralized beam.

5.1.5 Interface Design

Figures 5-11 and 5-12 show this interface in the elevation and plan views. In Stage I there are three positive ion neutral beam injectors, two deuterium and one tritium. For Stage II there are four negative ion NB injectors, two deuterium and two tritium. These injectors perform the function of plasma fueling as well as heating and current drive. The injectors are positioned in alternate sectors to facilitate maintenance. In both cases, the deuterium beams inject tangential (45°) to the plasma centerline. This increases the current drive efficiency for these beams. For the tritium high energy beams, the injection is tangential (36°) to the inner wall. Reducing the injection angle from 45° to 36° increases the penetration for the tritium while still providing good current drive efficiency. There is an excess of current drive capability in Stage II, and sacrificing some current drive to improve penetration is acceptable. Stage I has the current drive capability maximized to achieve 4 MA plasma current with only 50 MW injected power.

PLAN VIEW

Fig. 5-11. Plan and elevation views show Phase I positive ion NB interface.

Fig. 5-12. Negative ion NB interface shows different injection angles.

Note that the injectors have similar foot prints; however, their drift duct size and shapes are different. For the positive ion NB, the drift duct measures 80 cm wide by 105 cm high. This passes six beams (2 stocks of 3) with the cross over in the center of the drift duct. This arrangement results in a spread of injection angles about normal of $\pm 2^\circ$. The drift duct for the D^- -based beam is 50 cm wide and 200 cm high. There are six highly elliptical beams accommodated by this duct. The beam cross over is at the drift duct center with an angular spread of $\pm 5^\circ$ about the nominal angle.

There are no neutron shutters in the drift duct because the injectors operate continuously; however there are two valves to allow maintenance on the beam line while maintaining the torus vacuum. In the event of a failure requiring removal of the NBI to the hot cell, the double valve system provides tritium containment for both the NBI and the torus.

5.2 ELECTRICAL POWER SYSTEMS

The power handling and conversion system consists of the ac power system and the power supplies. The ac power system provides 60 Hz electrical power at the desired voltages to FED-R and to the supporting facilities. It includes facility substations, feeders, transformers, and circuit breakers. The power conversion system consists of rectifier inverter power supplies for converting ac and dc, and vice versa. Coil protection includes instrumentation and control, dump resistors, and dc circuit breakers.

5.2.1 AC Power Systems

The ac power system provides all 60 Hz electrical power for operating FED-R and its supporting facilities. Very large pulsed power supply load that exists during startup and shutdown is supplied by flywheel energy storage units. Load that exists throughout the burn phase is supplied directly from utility lines.

Two utility power lines are proposed; one to supply pulsed power to variable loads; and one to provide steady power to nearly constant loads that need better voltage regulation. In addition, standby diesel

generators and uninterruptible power supplies are proposed for essential protection systems and for operating safety related equipment.

Requirements

The ac power system must provide the 60-Hz pulsed and steady-state power for the FED electrical loads and the facility support loads identified in Tables 5-7 and 5-8, respectively. This system also provides the backup power for essential loads and noninterruptible loads. Fault protection is provided by coordinated circuited breakers in each feeder line.

Referring to Table 5-7, the TF coils require nearly half the total power. Because of the very high load requirements and the need to limit the short circuit current with high impedance transformer, two substations are proposed. Each substation has two 3 ϕ , 13.8 kV circuits rated 120/160/200 MVA. One substation, having the better voltage regulation, supplies the TF coil power supplies and the facility loads. The other substation provides power to the remaining loads.

Table 5-8 is an estimate of the facility power loads and identifies the essential loads transferrable to the diesel generators when loss of utility power occurs. Superconducting coil protection and environmental safety related equipment will be provided with uninterruptible power.

Table 5-9 identifies the utility line and backup power requirements. Two power lines are proposed, each having line voltages ≥ 230 kV. Power transfer between substations to selected loads will also be provided. Diesel generators provide about 4 MW to essential loads during utility power failures.

5.2.2 Power Conversion Systems

The power conversion for the TF coils and PF coils is described in this section.

Table 5-7. FED-R quasi-steady electrical power loads

	Stage I MW	Stage II MW
TF coil power supplies ^a	200	310
Water-cooled PF-coil power supplies ^b	57	88
NB injector power supplies	140	140
MGF drive motors ^b	4	4
Compressor motors	5	10
Water coolant pumping	6	15
Fuel processing (decont. and isotope separation)	5	8
Vacuum pumping	2	2
Helium cooling pumping (first wall)	4	18
Facilities (excluding any of above loads)	12	12
Total	435	607
	% total	% total
TF coils	46	51
PF coils	13	15
Neutral beams	32	23
Pumping systems	3	6
Miscellaneous	6	5
Total	100	100

^a Includes inductive power (~10%) at beginning of pulse.

^b Averaged over pulse duration.

Table 5-8. Estimate of facility power loads*

Load description	Power	
	kW	kVA
Building and ground lighting	600	600
HVAC for all buildings	3200	4000
Fuel processing	5000	6000
Shop and service equipment	2500	3000
Laboratory and office equipment	900	1000
Diagnostics	1500	1800
Data acquisition	90	100
Process C&I ^a	90	100
Emergency tritium removal (1) ^a	1300	1400
(2) ^a	1300	1400
Emergency cooling ^a	500	600
Essential lights ^a	50	50
Uninterruptible power ^a	40	50
• Essential communication and displays (10)		
• TF coil protection (10)		
• PF coil protection (10)		
• Fuel recovery (5)		
• Safety related valves (5)		

* No appreciable difference between Stage I and Stage II.

^aTransferable to diesel generator plant.

Table 5-9. Utility line and backup power requirements

I. Two independent utility power lines

Steady power line

Capacity	400 MVA
Voltage	<u>>230 kV</u>
Regulation	<u>+5% daily variation</u> <u>+1% normal transient</u>
Major loads	
Facility	20 MVA
TF power conversion	340 MVA

Pulsed power line

Capacity	500 MVA
Voltage	<u>>230 kV</u>
Regulation	<u>+10% daily variation</u>
Major loads	
Neutral beam power	160 MVA
PF coil power	120 MVA
Coolant pump power	40 MVA
Compressor power	40 MVA

II. Backup power

Diesel generators

Number	2
Capacity per generator	2.5 MVA
Voltage	2.4 kV
Regulation	<u>+5%</u>

Essential loads

Emergency tritium removal	2800
Emergency cooling	600
Essential lights	50
Process I&C	100
Uninterruptible power supplies	40

TF coil power conversion

The FED-R TF field is provided by 12 water-cooled copper coils. The coils are charged in ten minutes and operate in steady state for long periods of time. Toroidal field coil power conversion requirements data are given in Table 5-10. The TF coils must carry equal currents during charge, steady state regulation, and discharge to prevent high unbalanced forces on the coils. For this reason the TF coils are connected in series. The coils require about 360 kA during Stage II operation.

To provide 360 kA, six 60 kA power supply bays are connected in parallel. Each of the six power supplies consists of twelve power units, as shown in Fig. 5-13. Each power supply contains the equipment listed in Table 5-11. The TF coil power conversion requires 72 power modules; 72 rectifier transformers; 72, 13.8 kV circuit breakers; and large quantities of bussing and cables. Air-cooled aluminum bussing connect the power supplies to a common water-cooled copper bus at the entrance to the reactor building.

Only two of the twelve power modules of a TF coil power supply have thyristor rectifiers for current regulation. The auto-transformer provides coarse step adjustment of the current. The thyristors are controlled to provide the fine adjustment and current regulation.

Power conversion for the PF coils

The PF coil power supplies must be designed and controlled to provide the current waveforms shown in Fig. 5-14. For Stage II operation, the maximum current of the normal water-cooled coils is 120 kA, and the maximum current of the superconducting coils is 50 kA. The voltage of the PF coil power supplies depends on the inductance and resistance of the coils as well as the current and rate of change in current. Table 5-12 is the inductance/resistance matrix for the FED-R coils defined in the design control parameters (see appendix). From these data, the maximum voltages and currents of the power supplies were calculated and listed in Table 5-13.

Table 5-10. TF coil power converter and protection requirements data

Parameter	
Type of coil	Water-cooled copper
Number of coils	12
Charge time	600 s
Steady state coil operating current	300 (Stage I) kA 360 (Stage II) kA
Steady state current regulation	$\pm 0.5\%$
Total stored energy	3.6 (Stage I) GJ 5.6 (Stage II) GJ
Coil resistance during operation	1.80×10^{-4} ohm

Fig. 5.13. One of six TF coil power supplies that is connected to a common copper bus at the entrance to the reactor building of FED-R.

Table 5-11. TF coil power supply data for FED-R

Number of 870 V.D.C., 60 kA power supply bays	6
Number of power supply bays needed for Stage I operation	5
Number of 12 pulse, 5 kA, 870 V.D.C. power units/power supply	12
Number of 3 winding rectifier 5 MVA transformers per power supply (13.8 kV/650 V/650 V)	12
Number of 13.8 kV circuit breakers per power supply	12
Number of 13.8 kV, 60 MVA tapped auto transformers per power supply	1
Mean length of 3" x 48" aluminum bus per power supply, meters	200
Total length of 2.5" x 72" water-cooled copper bussing, meters	200
Mean length of 15 kV, 500 MCM cable per power supply, meters	1000
Mean length of 1 kV, 500 MCM cable per power supply, meters	2000

Fig. 5.14. - PF coil current waveforms.

Table 5-12. PF coil inductance/resistance matrix

Inductance in millihenries						
Coil	1-i	2-i	3- ϕ	4- ϕ	OH	Plasma
1-i	2.53	0.306	1.03	2.21	0.648	0.0295
2-i	0.306	29.6	2.39	23.7	1.45	0.0837
3- ϕ	1.03	2.39	57.8	36.9	0.823	0.106
4- ϕ	2.21	23.7	36.9	1023	3.36	0.404
OH	0.648	1.45	0.823	3.36	21.1	0.0436
Plasma	0.0295	0.0837	0.106	0.404	0.0436	0.0064
Resistance in milliohms						
Coil resistance	0.61	2.79	0	0	7.66	

Table 5-13. PF coil power supply data

Coil	Maximum ratings		MVA
	Kiloamps	Volts	
1-i	120	120	20
2-i	120	600	80
3- ϕ	50	220	15
4- ϕ	50	2500	180
OH	120	1400	240

The OH power conversion equipment consists of two 60-kA power supplies connected in parallel to a cylinder-operated DPDT switch, as shown in Fig. 5-15. The switch reverses the polarity of the power supplies relative to the OH coil when the current passes through zero. The power supplies contain twelve, 5-kA thyristor rectifiers to provide the desired current waveform given in Fig. 5-15. Table 5-14 identifies principal components and provides data for the OH coil power conversion equipment.

The OH coil is made of water-cooled copper windings requiring large peak power. The coil current is reduced to zero within 100 seconds after the startup is completed. This pulse of power is taken from the MGF energy storage units.

The outside superconducting EF coils also require large pulses of power during startup only. Their power supplies are connected to the MGF units as shown in Fig. 5-16 for a typical superconducting coil power supply. The pulsed power for startup passes from the MGF units through the rectifier power modules. The power supply for the low voltage burn operates from the utility line. DC circuit breakers disconnect the power supply if a coil quench occurs. The energy in the coil is discharged through dump resistors that could be located on the top of the building. Only one power supply bay per superconducting coil is needed to supply 50 kA.

The inside EF coils are water-cooled copper coils that operate at full current during the burn phase of the cycle. Therefore, the power converters for these coils are connected to the utility power line as shown in Fig. 5-17. This figure shows one of the two power supplies needed for the top inside coil. A single transformer with two secondary windings provides 2.4-kV power to both power supplies of this coil, only one of which is shown in the figure. The 2.4-kV power reduces the cost of the 24 circuit breakers and 24 rectifier transformers.

The lower inside EF coil requires about 70-MW peak power; therefore, the power conversion units are supplied with 13.8-kV power. Power supplies for this coil are similar to the ones for the OH coil (Fig. 5-17) except that there is no DPDT polarity reversing switch and the power supplies are connected to the utility power substation, not the MGF units.

Fig. 5.15. One of two power supplies for the OH coil of FED-R.

Table 5-14. OH coil power supply data for FED-R

Number of 1400 V.D.C., 60 kA power supply bays	2
Number of power supply bays needed for Stage I operation	2
Number of 13.8 kV circuit breakers per power supply	12
Number of 3 winning, 5 MVA, rectifier transformers per power supply (13.8/1.05/1.05 kV)	12
Number of 12 pulse, 5 kA, 1400 V.D.C. power units per power supply	12
Number of 1400 V.D.C., 120 kA, DPDT cylinder operated switches	1
Total length of 3" x 48" aluminum bussing, meters	400
Total length of 2" x 30" water-cooled copper bussing, meters	50
Length of 15 kV, 500 MCM cable, meters	1000
Length of 3 kV, 500 MCM cable, meters	2000

Fig. 5.16. Typical power supply for FED-R superconducting coils.

Fig. 5.17. One of two power supplies for the top inside EF coil of FED-R.

5.2.3 Electrical Energy Storage System

The short duration pulsed loads of the superconducting coils and the OH coil are connected to the electrical energy storage system to reduce the peak power demand on the utility line. The energy storage and peak power of these coils are given in Table 5-15.

The proposed energy storage system consists of two MGF units, each rated 300 MVA and having 3.2 GJ of available stored energy. The pertinent design data for the motor and generator are given in Table 5-16. The MGF units are connected to the power modules of the OH coil power supplies and superconducting coil power supplies through current limiting reactors, as shown in Fig. 5-18. The coil power supplies can also be connected to the utility line with the MGF by-pass switch so that maintenance and checkout can be performed with the MGF units shut down. The number in parenthesis shown next to the current limiting reactors corresponds with the number of power modules in the respective power supplies, i.e., there are a total of 44 current limiting reactors and a total of 44 circuit breakers.

5.3 DIAGNOSTICS, INFORMATION, AND CONTROL SYSTEMS

Diagnostics, information, and control include the functions and processes for complete operation of the machine and facility. Diagnostics is defined to include all instruments that provide data related to the properties of the plasma. These can further be subdivided into those instruments needed to operate a reactor and those needed to study specific characteristics of the plasma.

The FED-R will be provided with only those diagnostics or sensor necessary for machine operation, performance verification, and safety.

FED-R will have several phases of operation, as shown in Table 5-17. Special attention is required for instruments related to the first wall and test modules. Diagnostics must provide the information required for feedback control such as plasma current, distribution, position, density, neutron yield, and temperature.

Table 5-15. MGF energy storage requirements data

	Stage I	Stage II	Units
OH coil			
Startup energy	1.4	1.4	GJ
Shutdown energy	3.6	3.6	GJ
Peak startup power	240	240	MVA
Peak shutdown power	25	25	MVA
Top superconducting coil			
Startup energy	0.06	0.1	GJ
Shutdown energy	0.06	0.1	GJ
Peak startup power	10	15	MVA
Peak shutdown power	10	15	MVA
Bottom superconducting coil			
Startup energy	0.8	1.3	GJ
Shutdown energy	0.8	1.3	GJ
Peak startup power	110	180	MVA
Peak shutdown power	110	180	MVA
MGF requirement			
Total stored energy	5.8	6.4	GJ
Maximum power	360	435	MVA

Table 5-16. Energy storage system design data

Parameter	Value	Units
A. Drive motor		
Type	Wound rotor induction	
Quantity	2	
Shaft power	10,000	HP
Frequency	50/60	Hz
Phase	3	
Stator voltage	13.8	kV
B. Generator		
Type	Synchronous	
Quantity	2	
Peak load capacity	300	MVA
Duty cycle	2	%
Voltage	13.8	kV
Phase	3	
Voltage regulation	+1.0	%
Base frequency	50/60	Hz
Maximum frequency	1.5 × base	Hz
C. Stored inertia energy per MGF unit		
100% speed	5.8	GJ
67% speed	2.6	GJ
Available energy/MGF unit	3.2	GJ

Fig. 5.18. One-line diagram of the AC energy storage system for FED-R.

Table 5-17. FED-R phases of operation

Subsystem checkout**System checkout**

- Hydrogen operation
- Feedback control studies
- Power supply tests
- Leak detection and localization

Hydrogen plasma

- Initial operation
- Performance verification

Deuterium plasma

- Instrument change
- Performance verification

D-T plasma

- Performance verification
- Optimization of operating parameters

Nuclear testing

- Diagnostics for control
- Diagnostics for reestablishing performance
- Engineering testing

Because of the radiation environment, all instrumentation required for operation must be designed for remote maintenance and calibration. One torus sector will be dominated by plasma diagnostics, but it is expected that there will be some diagnostic instrumentation in other sectors. The plasma diagnostic instrumentation must supply inputs to the control system since a plasma of repeatable and steady characteristics can only be obtained with a workable control system. Control is achieved in a manner characteristic of present-day tokamak operation except for the plasma current which is driven by the beams in FED-R. The subsystems being controlled are the PF coil biasing, beam injection power, and fueling rate.

The design of FED-R will require a reliable and available control system; therefore, redundancy will be used for all of the control instruments. For instance, two sets of $\dot{B}$ loops (to monitor the plasma current) will be constructed, with the second set being available if there is a failure of the first set.

The major fraction of diagnostic instruments is presently used by physicists to measure progress in improving the plasma performance as input parameters are varied. They are also used to understand unexpected deficiencies in performance. Because comparison between theory and experiments invariably involves the subtleties of plasma profiles, more detailed diagnostic information is needed for analysis than for control. FED-R will not be provided with the analysis information diagnostics. Therefore, the space and costs required are much lower than for the FED Baseline.^{1,2}

Information and control system requirements comprise the hardware and software to perform all programmable processes for FED-R. These include control, performance and status monitoring, data acquisition and processing, analysis, display, and archiving. A control center, local and remote controls, timing and communications, and a safety system are required.

REFERENCES

1. K. W. Ehlers and K. N. Leung, "Self-extraction H^- or D^- Ion Source," presented at the 24th Annual Meeting of the APS Division of Plasma Physics, New Orleans, La. (Nov. 1982).
2. O. A. Anderson, "The TFF Accelerator for Fusion Reactor Injection," *ibid.*
3. W. S. Cooper, O. A. Anderson, D. A. Goldberg, and J. Fink, "Improved Negative-Ion Based Neutral Injection System," *ibid.*
4. M. W. McGeoch, *Proc. 2nd Int. Symp. on the Production and Neutralization of Negative Hydrogen Ions and Beams*, Brookhaven National Laboratory, Upton, N.Y. (1980), p. 304.
5. P. T. Kirstein, G. S. Kino, and W. E. Waters, *Space-Charge Flow*, McGraw-Hill Book Co., New York (1967), p. 255.
6. H. A. C. Hogg, "Periodic Electrostatic Beam Focusing," *Proc. IEE*, Part B, Suppl. 10-12, p. 1016 (1958).
7. K. W. Ehlers, K. N. Leung, "Characteristics of a Self-Extraction Negative Ion Source," *Proc. 2nd Int. Symp. on the Production and Neutralization of Negative Hydrogen Ions and Beams*, Brookhaven National Laboratory, Upton, N.Y. (1980), p. 198.
8. J. R. Pierce, *Theory and Design of Electron Beams*, Van Nostrand Co. (1954), p. 41.
9. J. H. Fink, *Evaluation of Laser Stripped Negative Ions as a Source of Neutral Beams for Magnetic Fusion Reactors*, Lawrence Livermore National Laboratory, UCRL-87301 (1982).
10. K. N. Leung and K. W. Ehlers, "Self-Extraction Negative Ion Source," *Rev. Sci. Instrum.* 53 (1982) 803-6.
11. R. M. Gilgenbach, et al., "Electron Cyclotron/Upper Hybrid Resonant Pre-Ionization in the ISX-B Tokamak," *Nucl. Fusion* 21 (1981) pp. 319-327.
12. D. J. Grove, "The Status of TFTR," in *Proc. Fourth Topical Meeting Technology of Controlled Nuclear Fusion*, U.S. Dept. of Energy CONF-801011 (1981), pp. 957-960.

6. NUCLEAR SYSTEMS DESIGN

Emphasis is placed on two major components of the nuclear system, viz., the vacuum vessel/first wall and the bulk shield. In the design of the vacuum vessel/first wall, emphasis is placed on the outboard wall of the vacuum vessel which must be compatible with the test modules located just outside of that wall.

The bulk shield is addressed primarily as it is influenced by the shutdown dose requirements for hands-on maintenance activities.

6.1 FIRST WALL AND VACUUM VESSEL DESIGN

There are two principal nuclear test-related requirements in the definition of the vacuum vessel. These are as follows:

- Maximize the uncollided 14 MeV neutron flux through the outboard wall into the test modules.
- Provide a configuration which permits test module replacement without opening up the vessel to atmosphere.

The first requirement above had a significant effect on the selection of wall materials, both structural and coolant. It also resulted in a careful placement of structural members to assure that the structure (while adequate) did not compromise the neutronic performance unduly.

The second requirement above resulted in a selection of the vacuum boundary at the first wall location. This location favors the objective of rapid test module replacement but results in some reduction in maintenance access to the other parts of the vacuum vessel, such as the inboard wall. An advantage of this configuration is that all bulk shielding is outside of the vacuum boundary and, hence, coolant leaks do not affect plasma operation.

6.1.1 Requirements and Design Features

There are two stages of D-T operation planned for FED-R nuclear testing. These stages are described in detail in Appendix A. Stage I has a neutron wall loading of 0.4 MW/m^2 and lasts six years. A higher

neutron wall loading of 1.3 MW/m^2 exists in Stage II operation which lasts 9.5 years. The spatially averaged plasma radiation heat load to the first wall is 10 W/cm^2 during Stage I and 25 W/cm^2 during Stage II. A bare stainless steel inboard, top, or bottom surface of the vacuum vessel could experience significant melting and vaporization due to disruption heating, and thus have a limited life. Therefore, graphite tiles are provided on these first wall surfaces. These tiles are expected to last the lifetime of the device. According to the present understanding of plasma disruption phenomenon, the outboard surface in front of the test modules will not experience severe disruption heating conditions and, therefore, is designed to be an actively cooled "thin" bare stainless steel wall.

The disruption armor tiles are mechanically attached in a bolted arrangement (similar to that presented in Refs. 1 and 2) to the shield surface. Since the thermal contact at this interface will be poor, the tiles were assumed to be cooled by radiation only. No conduction heat transfer from the tiles to the shield was included in the thermal analyses. The heat loads on the outboard first wall due to plasma radiation, radiation from the disruption armor, and nuclear heating are listed in Table 6-1. Based on results for 5 cm thick graphite armor tiles, the bare outboard stainless steel experiences a radiation heat flux of 35 W/cm^2 from the armor tiles during Stage II, which along with the plasma thermal flux (mainly radiation) results in a total surface heat flux of 60 W/cm^2 . Nuclear heating was also included in the thermal analyses.

Neutral beam shine-through during Stage I operation results in a 6 W/cm^2 heat load to the outboard first-wall panels. The total surface heat load during Stage I operation on the regions subjected to shine-through is therefore 27 W/cm^2 , including radiation from the plasma and inboard armor tiles. The additional shine-through heating can be tolerated by the outboard panels since they are designed to operate with a Stage II surface heat load of 60 W/cm^2 .

Since the key goal of FED-R is to provide a test bed for all types of fusion neutron application modules, high transmission of neutrons through the outboard first wall and into the test modules is a key

Table 6-1. Operating heat loads for FED-R first wall

	Stage I	Stage II
Heat load from plasma ^a - W/cm ²	10	25
Heat radiated to outboard first wall from 5-cm thick inboard tiles - W/cm ²	11	35
Nuclear heating - W/cm ³		
• Graphite	2.8	8.4
• 316 stainless steel	4.8	14.4
• Aluminum (optional wall material)	2.7	8.0
Neutron wall load - MW/m ²	0.4	1.3

^aIncludes radiation, charged particles, and neutral atom bombardment and takes into account divertor effectiveness.

performance objective. Various design concepts and materials were compared on the basis of relative transmission of 14 MeV neutrons. Future evaluations of the first wall should include neutron transmission requirements and performance of the first wall in terms of the entire neutron energy spectrum.

6.1.2 Baseline Description

The first wall and vacuum vessel is segmented into six 60° toroidal sectors. Each sector has individual coolant manifold lines and is welded to adjacent sectors.

The vessel is constructed of 316 stainless steel double-skin ribbed panels. Helium is employed as coolant in the outboard wall region. Water is the coolant in the inboard, top and bottom wall regions. The ribbed panel configuration and dimensions are shown in Fig. 6-1. Panels are fabricated by machining the 4.5-cm thick coolant passages into the plate on the plasma side and then electron beam welding the back plate. Front and back skins are both 0.24 cm thick, resulting in a total wall thickness of approximately 5 cm.

Welds between sectors are shielded against excessive neutron damage and from direct contact with the plasma. The weld joint configurations for the inboard, top, bottom and outboard vessel surfaces are shown in Figs. 6-2, 6-3, and 6-4, respectively. Double welds are provided at all locations. Inboard and top (and bottom) vessel surfaces are welded (Figs. 6-2 and 6-3) from the inside of the device prior to installation of the outboard surface. The outboard surfaces are welded (Fig. 6-4) together from the outside of the vessel. The configuration of the welded surfaces which form the transition from the inside welds on the top and bottom surfaces to the outside welds on the outboard surface is shown in Fig. 6-5.

Helium is the coolant for the outboard first wall surfaces directly in front of the test modules. Helium was selected because it provides a higher neutron transmission compared with water. Water coolant is provided for the remaining inboard, top, and bottom surfaces. Neutron transmission is not important in these regions, and the required pumping

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Fig. 6-1. Vacuum vessel wall in front of nuclear test modules in FED-R.

Fig. 6-2. Inboard wall joint.

Fig. 6-3. Top wall joint.

Fig. 6-4. Outboard wall joint.

Fig. 6-5. Assembled top and outboard wall joints.

power for water is much lower than that required for helium coolant. First wall coolant system performance parameters for Stage II are listed in Table 6-2 for both coolant loops.

The baseline outboard first wall design allows 82 percent of the 14 MeV neutrons to be transmitted to the test modules without interaction with first wall structure or coolant. This result depends largely upon using helium coolant. For water coolant, only 55 percent of the neutrons would pass through the first wall unattenuated.

Consequences of loss of plasma

In the event of a rapid quench of the plasma, such as by a major disruption of the plasma current, an electronic system with sensors monitoring the plasma density will cut off the neutral beam voltage before any damage to the first wall occurs. A backup electronic system with a sensor monitoring a different parameter, such as the plasma current, will cut off the beam voltage in event of failure of the primary cutoff system.

The installation of protective plating would have a serious attenuation effect on fusion neutrons. There seems little point in providing protective plating for the long-pulse operation of FED-R, since it is unlikely that the beams can be manually shut down within the few seconds necessary to burn through the plating. Therefore, efforts should concentrate on providing an electronic fail-safe system for shutdown of the beams.

In the unlikely event of failure of both the primary and backup cutoff systems, the first layer of the vessel double wall may be burned through, resulting in a substantial helium influx into the torus. The increase in gas pressure in the torus will be adequate to stop the beam before burn-through of the rear layer of the wall. However, a repair to the wall would have to be made.

6.1.3 Alternate Concepts

Several alternate vacuum vessel and coolant materials were examined. In addition to 316 stainless steel, aluminum (Al 5083-0) was considered

Table 6-2. Base coolant system performance, Stage II

Location of Cooled Surfaces	Outboard	Inboard, top, and bottom
Coolant	Helium	Water
Inlet temperature -°C	60	60
Outlet temperature -°C	96	80
Inlet pressure -MPa (psi)	2.0(300)	0.7(100)
Pressure drop -MPa (psi)	0.057 (8.5)	0.01 (1.5)
Flow velocity in coolant passages - m/s	130	1.6
Flow rate - kg/s	424	900
Pumping power - MW	8.7	0.01

for the vessel structure. Aluminum offers the advantages of higher thermal conductivity (resulting in lower operating temperatures and thermal stresses), higher transmission of 14 MeV neutrons for the same thickness, and higher electrical conductivity. Disadvantages of aluminum compared to 316 stainless steel are lower strength, lower allowable temperature, and less reactor relevancy (due to its low allowable temperature).

The three coolant materials considered were helium (gaseous), water, and sodium potassium (NaK 22/78). The latter is often used in radiation test facilities and certain 14 MeV point sources. Helium and NaK allow a higher fraction of the 14 MeV neutrons to pass through the first wall compared to water. Disadvantages of helium compared to water are that to obtain comparable heat transfer characteristics, helium normally operates at a higher pressure and results in larger pumping powers. NaK has disadvantages of high MHD pressure drops and pumping powers caused by flow across a magnetic field. NaK also imposes serious safety problems.

Thermodynamic, structural, and neutronic performance for these alternate concepts are discussed below. The rationale for selection of the baseline design is also discussed.

Rationale for baseline concept selection

Both stainless steel and aluminum first wall structures can be designed to have acceptable thermal and structural performance. There are only minor differences in the 14 MeV neutron transmission of a first wall designed for either of these materials, because an aluminum wall must be thicker. Since stainless steel is more reactor relevant, it was selected as the baseline first wall structural material.

A summary of the key thermal hydraulic, structural, and neutron transmission performance parameters are listed in Table 6-3 for the three coolant materials considered. It is clear that NaK is relatively unattractive, and hence not a viable candidate. The pumping power for the helium coolant flow system (8.7 MW at the reference coolant conditions) is many times larger than that for a water coolant system (0.2 MW);

Table 6-5. FED-R first wall coolant flow system conditions

- 316 SS temperature $\leq 400^{\circ}\text{C}$
- Coolant inlet temperature = 60°C
- Outboard wall surface heat load = 60 W/cm^2 (Stage 1I)

Condition	Coolant		
	Water	Helium	NaK
Inlet pressure - MPa (psi)	0.7 (100)	2.0 (300)	11 (1600)
Pressure drop - MPa (psi)	0.06 (9)	0.06 (9)	10.6 (1540)
Pumping power - MW	0.2	8.7	60
Minimum 316 SS thickness, t - cm (rib spacing held constant)	0.2	0.24	0.8
14-MeV neutron transparency; of first wall structure (316 SS) and coolant - %	55	82	65

however, this power is still relatively modest compared with that consumed by other reactor components such as magnets (less than 3 percent of the magnet power) and neutral beams. Based on the neutron transmission advantages of helium (about 82 percent for helium compared with about 55 percent for water) it was selected as the baseline coolant.

Several methods can be used to reduce the helium pumping power. Water cooling could be used in regions not requiring high neutron transmission. As pointed out in the thermodynamic performance section, the pumping power can be reduced from 9 MW to 2 MW if the maximum allowable stainless steel temperature is raised from 400°C to 500°C.

Figure 6-6 combines the results of thermal hydraulic performance (Fig. 6-7) and neutron transmission performance (Fig. 6-8). This figure demonstrates the significance of the trade-off between helium pumping power and neutron transmission for both stainless steel and aluminum walls. For stainless steel, the minimum helium pumping power of 6.5 MW occurs at a transmission of 78 percent and corresponds to a helium pressure of 4.0 MPa (600 psi). At lower pressure, the neutron transmission is higher because a thinner structure can be used. However, because of the decreased thermal inertia and convection heat transfer of the coolant at low pressures, the flow rate and pumping power must be increased to provide adequate cooling. At higher pressures, the wall must be made thicker, which results in lower neutron transmission. The thicker wall also requires a higher flow rate and pumping power to maintain the heated surface of the stainless steel below 400° C. For a stainless steel wall and using the conditions considered in these studies, both transmission and pumping power are adversely affected if the helium pressure is increased above 4.0 MPa (600 psi).

As shown in Fig. 6-6, the pumping power for an aluminum wall is larger than that for a stainless steel wall for transmissions above 77 percent. At the reference design transmission of 82 percent, the pumping powers are 8.7 MW for a stainless steel wall and 18 MW for an aluminum wall.

The 14 MeV neutron transmission of a first wall with a NaK coolant system is 65 percent for stainless steel and 70 percent for aluminum based on a required inlet pressure of 11 MPa (1600 psi).

Fig. 6.6. Pumping power as a function of 14 MeV neutron transmission through the vacuum vessel.

Fig. 6-7. Pumping power vs coolant pressure for aluminum and stainless steel vacuum vessels. Thermal load = 60 W/cm².

Fig. 6-8. Transmission of 14 MeV neutrons through aluminum and stainless steel vacuum vessel walls.

6.1.4 Thermodynamic Performance

Normal operating heat loads for the outboard first wall are summarized in Table 6-1. The Stage II surface heat flux of 60 W/cm^2 was used for thermal analyses of both stainless steel and aluminum outboard first walls. Water coolant was used for the structural material studies. Maximum stainless steel and aluminum outboard first wall temperatures for a range of thicknesses (distance from heated surface to coolant) are shown in Fig. 6-9. Thickness limits were determined based on maximum operating temperature limits of 400°C for stainless steel and 200°C for aluminum. As indicated in Fig. 6-9, to maintain the heated surface below these limits the thicknesses must be less than 0.66 cm for stainless steel and 0.85 cm for aluminum. Temperature differences from heated surface to cooled surfaces (ΔT) are also indicated in Fig. 6-9. These temperature differences were used to calculate the thermal stresses. Thermal stress limits are discussed in the structural performance section.

The configuration of the outboard first wall coolant flow passages is shown in Fig. 6-10. Assumptions used for the outboard first wall coolant flow analyses are listed below:

Inlet Temperature = 60°C

Rib Spacing (S) = 3.5 cm

An inlet temperature of 60°C for the primary coolant loop represents a reasonable outlet temperature from a heat exchanger connected to a secondary loop which exchanges heat with the environment.

The internal rib spacing (S) must be as large as possible to minimize the attenuation of the 14 MeV neutron. However, if the rib spacing is too large then the skin thickness required to maintain an acceptable bending stress becomes large, and a large fraction of the 14 MeV neutrons are excessively attenuated by the wall. A rib spacing of 3.5 cm results in a reasonable compromise between attenuation due to rib spacing and skin thickness.

Skin thicknesses (t in Fig. 6-10) of 0.20 cm for stainless steel and 0.26 cm for aluminum at a pressure of 0.7 MPa (100 psi) were used for these analyses. The skin thickness is proportional to the square root

Fig. 6.9. Outboard first wall temperatures.

Fig. 6-10. Vacuum vessel wall.

of the coolant pressure (for a constant rib spacing). Therefore, the thickness (t) was increased according to this relationship as the coolant pressure was increased above 0.7 MPa (100 psi).

The skin spacing (flow passage width, w) required for structural stability (discussed in the structural performance section) was varied as a function of skin thickness. For all three coolant materials, the optimum coolant passage width from a thermal hydraulic viewpoint was smaller than that required for structural stability.

A convection heat transfer coefficient of about $1.0 \text{ W/cm}^2\text{K}$ was required for the water coolant flow system to inhibit subcooled boiling at a pressure of 0.7 MPa (100 psi). The resulting pressure drop and pumping power for cooling the outboard wall are 0.07 MPa (0.1 psi) and 0.2 MW.

The sensitivity of helium coolant flow system pumping power to inlet pressure is shown in Fig. 6-7. The helium coolant system is sized to maintain the stainless steel temperature below 400°C and the aluminum temperature below 200°C .

At high pressures, good transport properties of helium make the required coolant velocity small, resulting in low pumping power. However, the skin must be thick at high pressures, resulting in a low coolant side structural temperature (for the heated surface temperature held constant) and thus requiring high coolant velocity and large pumping power. These competing influences result in a minimum pumping power at an intermediate pressure. The pressure at which this maximum pumping power occurs is very sensitive to the thermal conductivity of the wall. For a stainless steel wall, a minimum helium pumping power of 6.5 MW is achieved at an inlet pressure of about 4 MPa (600 psi). The required stainless steel wall thickness at this pressure is 0.48 cm.

For helium coolant pressures less than about 6 MPa (900 psi), the stainless steel wall requires less pumping power than the aluminum wall. The minimum helium pumping power for an aluminum wall occurs at a pressure greater than 20 MPa (3000 psi), which is an impractically high operating pressure regime. Since aluminum has a high thermal conductivity compared to stainless steel (about a factor of 10 higher), the

effect of the temperature drop through the wall thickness on pumping power is insignificant until pressures are high. Therefore, at high pressures (greater than 6 MPa) the pumping power for an aluminum wall is less than that for a stainless steel wall.

In addition to these thermal hydraulic considerations, the effect of operating pressure on wall thickness and the resulting 14 MeV neutron transmission must also be considered. The tradeoff between helium pumping power and neutron transmission is discussed in later paragraphs.

Helium pumping power was found to be very sensitive to the maximum allowable stainless steel temperature. Results of this sensitivity study are shown in Fig. 6-11 for several coolant pressures. If the maximum allowable stainless steel temperature is increased from 400°C to 500°C, the pumping power for a 1 MPa (150 psi) inlet pressure is reduced from 21 MW to 6 MW. However, the radiation damage rate is expected to be much higher at 500°C.

Sodium-potassium (NaK) was eliminated as a design option. For sodium-potassium (NaK) coolant, the pressure drop and pumping power are 10.6 MPa (1540 psi) and 60 MW. This pumping power is about the same as the total thermal power incident on the outboard first wall. The large pressure drop and pumping power are caused by the MHD effects of flowing a liquid metal across a magnetic field. This condition cannot be avoided since the flow must either cross the poloidal field of 0.9 T at the outboard wall (vertical flow direction) or the toroidal field of about 8 T (horizontal flow direction). Either flow direction also crosses the radial field of 0.4 T. The vertical flow direction was assumed for these analyses. There are also safety and maintenance concerns associated with using NaK, and the thermal hydraulic performance of this coolant is not as good as either water or helium. NaK is also inferior to helium as a neutron transmission media.

6.1.5 Structural Performance

The FED-R outboard first wall shell must withstand external atmospheric pressure loads and must tolerate stresses caused by coolant pressure and surface heating. The shell configuration used in the

Fig. 6-11. Pumping power for stainless steel vacuum vessel.

structural evaluation is shown in Fig. 6-12(a). The shell is a double-wall, rib-stiffened cylinder. Vertical stiffeners are provided to take vertical loads (pressure on top and bottom of the torus) and to help stiffen the shell against general instability failure. The shell wall depth is established to provide the required stiffness of the shell between stiffeners. The skin thickness must be adequate to take the hoop compression loads caused by the one atmosphere external pressure, Fig. 6-10(b), and to take local bending stresses caused by the coolant pressure. Increasing skin thickness decreases both the hoop compression and the local bending stresses. However, increases in skin thickness result in increased thermal stresses. Therefore, both minimum and maximum structural skin thickness cutoffs are considered.

Both stainless steel and aluminum alloys were considered for the shell structure. Properties used for the alloys are given in Table 6-4.

Results of stability analyses for the two materials are shown in Figs. 6-13 and 6-14. The required effective structural shell thickness is determined from

$$p' = \frac{E\bar{t}^3(\pi^2/\alpha^2) - 1}{12 r^3(1 - \nu^2)}$$

where

p' = critical external pressure for shell buckling, psi,

$\bar{t}$ = effective thickness, in.,

r = shell radius, in.,

ν = Poisson's ratio (0.3),

α = angle between vertical stiffeners.

This equation assumes that the curved (top and bottom) edges of the shell are free and that the straight (vertical) edges at the stiffeners are simply supported, which are conservative assumptions. A factor of five was applied to the critical pressure calculated on an elastic basis. The resulting effective thickness is that which would be required

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Fig. 6-12. FED-R first wall shell configuration and loads.

Table 6-4. Properties of alloys considered for shell structure

Property	316 stainless steel (annealed)	5083-0 aluminum
Elastic modulus (E)	26×10^6 psi	10×10^6 psi
Yield strength	30 ksi	19 ksi
Coefficient of thermal expansion	17×10^{-6} in./in./°C	24×10^{-6} in./in./°C
Allowable stress (S_m)	20 ksi	13.7 ksi

Fig. 6-13. First wall depth and thickness requirements for stainless steel.

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Fig. 6-14. First wall depth and thickness requirements for aluminum.

for a solid thickness shell. These values are converted to the required moment of inertia (I_{reqd}) based on the formula

$$I_{\text{reqd}} = \frac{\bar{t}^3}{12(1 - \nu^2)} = \begin{array}{l} \text{moment of inertia per unit length} \\ \text{required to prevent buckling.} \end{array}$$

The required wall depth, h , (skin spacing) is then calculated using the following equation:

$$h = t + \sqrt{\frac{2(1 - \nu^2)}{t} I_{\text{reqd}} - \frac{t^2}{3}}$$

where

h = wall depth

t = required skin thickness (see Fig. 6-1).

Shown in Figs. 6-13 and 6-14 are combinations of depth and skin thickness with one intermediate stiffener. Results show that the shell depth can be reduced by providing the additional intermediate stiffeners. The tradeoff in this case then becomes whether less frequent stiffeners or a lower amount of shell material is most desirable from the neutron transparency standpoint. For the present study, more frequent stiffeners are preferred.

Shown on Figs. 6-13 and 6-14 are minimum skin thicknesses (t) to provide the required strength for combined hoop compression (one atmosphere) and local bending (150 psi coolant pressure). The stress due to the hoop compression is dependent only on the skin thickness. The stress due to the coolant pressure depends on both the skin thickness and rib spacing. The minimum values shown in the figures are based on a rib spacing of 3.0 cm. The allowable stress used for the combined primary and bending stress is $1.5 S_m$. The resulting desired configuration for the stainless steel shell consists of skin thicknesses of approximately 0.22 cm and a shell depth of approximately 5.0 cm. It is assumed that the rib thickness is the same as the skin thickness. For

the skin thickness of 0.24 cm, the rib spacing can be increased to 3.5 cm.

Thermal stresses were calculated for the stainless steel first wall shell with helium coolant. Results, along with stresses for hoop compression and local bending, are shown in Fig. 6-13. The critical location for combined compressive stresses is on the skin surface toward the plasma and at the rib joint. Maximum tensile stresses are about half of the total compressive values shown and occur on the inside of the skin, away from the plasma, and at the rib joint as shown in Fig. 6-15.

Fatigue life of the first wall shell is a function of the cyclic nature of the stresses. As shown in Fig. 6-15, the thermal stress increases with increasing skin thickness. At a thickness of 0.57 cm, the alternating thermal stress (1/2 peak) is 320 MPa (47 ksi), which corresponds to a life of 10^4 pulses. Thus, this thickness is considered to be an upper limit in the present design study. Cycling of the hoop compression and local bending stresses also affects the fatigue life. For combinations of stresses at the design thickness of 0.24 cm, the following fatigue lives are predicted:

Thermal stress	40,000 pulses
Thermal stress + bending	14,000 pulses
Total stress	7,000 pulses

6.1.6 Neutronic Analyses

The relative transmission of the 14 MeV neutron flux for the structural and coolant materials considered for FED-R are shown in Fig. 6-16. This analysis was performed with the MCNP Monte Carlo code. In the MCNP input, this flux group extends from 13.5 MeV to 15 MeV, so a few collided neutrons are also included in the results. The plot shows how neutron transmission deteriorates as the structure and coolant thicknesses increase. For example, assuming a first wall comprised of 0.6 cm of stainless steel and 4 cm of water, the 14 MeV neutron transmission will be $0.67 \times 0.58 = 0.50$ (50 percent of the 14 MeV neutrons will be transmitted without energy degradation).

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Fig. 6-15. Stresses in the stainless steel first wall shell.

Fig. 6-16. 14-MeV neutron transmission through first wall materials.

The internal ribs were included in the neutron transmission calculations by using a "smeared" thickness. The "smeared" thickness ($\bar{t}$) is given by:

$$\bar{t} = 2t + \frac{t_r w}{s}$$

where

t = skin thickness from heated surface to coolant (equal thickness assumed for back panel),

t_r = rib thickness,

w = coolant passage depth,

s = spacing between ribs.

Rib thickness (t_r) and spacing (s) were held constant at 0.2 cm and 3.7 cm, respectively. The coolant passage depth (w) is calculated as a function of the skin thickness (t) according to the structural stability requirements discussed in the previous section.

The 14 MeV neutron transmission is shown in Figs. 6-17 and 6-18 as a function of skin thickness for stainless steel and aluminum with water coolant. Structure is sized on the basis of structural stability, maximum temperature, thermal stress, and minimum gage limits. For both structural materials, increasing the skin thickness required for structural stability results in significant decreases in the water passage width (shown in Figs. 6-13 and 6-14). This leads to a higher neutron transmission. The maximum transmission occurs at the maximum allowable structural thickness, which is based on a thermal stress limit for stainless steel and a maximum temperature limit (200°C) for aluminum. Maximum transmissions for stainless steel and aluminum with water coolant are 55 percent and 58 percent, respectively.

Helium permits a 100 percent neutron transmission for the range of thicknesses considered. Therefore, for helium cooled walls the maximum transmission occurs at the minimum structural thickness. Minimum gages for stainless steel and aluminum at a helium coolant pressure of 0.7 MPa

Fig. 6-17. Vacuum vessel wall thickness as a function of neutron transmission.

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Fig. 6-18. Vacuum vessel wall thickness as a function of neutron transmission.

(100 psi) are 0.2 cm and 0.26 cm, respectively. The resulting neutron transmissions are 85 percent for stainless steel and 88 percent for aluminum. Increasing the helium coolant pressure requires the structural thickness to increase and thus decreases neutron transmission. The effect of increasing helium coolant pressure on the 14 MeV neutron transmission is shown in Fig. 6-8.

6.2 BULK SHIELD

6.2.1 Operating Requirements

Radiation criteria and shielding design for the FED-R concept are different in several respects from those for the FED Baseline. These differences arise from the following design features of FED-R:

1. Normal-conducting TF coils
2. Finger joints in the TF coils with a corresponding requirement for manned access
3. Emphasis on rapid replacement of test modules.

Since the coils are normal-conducting, resistive heating is far in excess of any contemplated nuclear heating. Also, the coil resistance is already high enough to make neutron-induced resistance increases unimportant. Hence, exposure limits are governed by insulation damage. For plate-type coils, inorganic insulation on which can tolerate large radiation exposures is employed. Therefore, the principal shielding problem for FED-R is controlling the shutdown dose rates around the machine and in the reactor building. An associated neutronics problem is maximizing the 14-MeV neutron transport into the test modules.

The above considerations lead to the FED-R shielding design criteria listed in Table 6-5.

6.2.2 Design Description

The shielding design approach for FED-R was governed by several groundrules:

1. Meet the criteria of Table 6-5;

Table 6-5. FED-R shield design criteria

Maximum lifetime dose to TF coil inorganic insulation	10^{11} rad
Maximum lifetime dpa to TF coil copper	No limit
Maximum nuclear heating rate in inboard TF coils	<20% of resistive heating
14-MeV neutron transmission through test module windows	Maximize
Shutdown dose rates	
General level in cell	5.0 mrem/hr at 36 hours
Local hot spots	<100 mrem/hr
Near finger joints	5 mrem/hr at 36 hours
Maximum at finger joint	20 mrem/hr at 36 hours

2. Use conventional, inexpensive, qualified materials such as water and steel;
3. Where possible, utilize the TF coils themselves as part of the bulk shield;^{*} and
4. Do not take undue credit for test module attenuation.

Exploratory shielding calculations were performed for several materials, both in homogenized and heterogeneous configurations. These analyses are briefly described in Section 6.2.3.

The resulting baseline shield design is shown in Fig. 6-19 and is described in Table 6-6.

The shielding arrangement in the vicinity of the finger joints requires further explanation. According to the nuclear analysis reported in Section 6.2.3, activation of the copper in the PF coil will result in a maximum shutdown dose rate of 20 mrem/hr at the finger joint, assuming no additional contributions from the shield, TF coil structure, or bolts. As shown in Fig. 6-20, shutdown gammas from the shielding and structure can be controlled by judicious use of lead cladding. However, possible activation of the bolts and cover plates remains a problem for further investigation.

6.2.3 Analysis

Exploratory calculations of the FED-R bulk shield were performed for a homogenized mixture of 90 percent stainless steel (316) and 10 percent borated water. The Los Alamos ONEDANT one-dimensional discrete ordinates code was used with a 30-neutron 12-gamma cross-section library. Output quantities consisted of neutron flux, dpa rate, insulation dose rate, and heating rate in both the inboard and outboard TF coils. For example, the insulation dose in the outboard coils is shown in Fig. 6-21, for an outboard shield thickness of 30 cm and no credit for the modules. The results would be little different if unborated water had been used.

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^{*} A mixture of 90 percent copper, 10 percent water is a very satisfactory fusion neutron shield.

Fig. 6-19. FED-R bulk shield.

Table 6-6. FED-R bulk shield description

Inboard Shield

Inboard wall loading	0.32 MW/m ² for 6×10^7 s 0.96 MW/m ² for 9.5×10^7 s
Shield thickness	45 cm
Shield composition	90% steel, ^a 10% water
Maximum nuclear heating at beginning of TF coils	0.1 W/cm ³
Insulation lifetime dose	5×10^{10} rad 4×10^{20} n/cm ²
Maximum lifetime copper dpa	0.1 dpa

Outboard Shield

Outboard wall loading	0.4 MW/m ² for 6×10^7 s 1.3 MW/m ² for 9.5×10^7 s
Shield thickness	
Portion between modules and coils	25 cm
Credit for module attenuation	<u>10 cm</u>
	35 cm
Intercoil shielding	<u>90 cm</u>
	125 cm
Shield composition	
First 25 cm	95% steel, ^a 5% water
Intercoil shield	Graded shield consisting of alternating layers of Nitronic-33 and water, clad by 5 cm lead
Maximum nuclear heating at beginning of TF coils	0.25 W/cm ³
Insulation lifetime dose	1.4×10^{11} rads 1.1×10^{21} n/cm ²
Maximum lifetime copper dpa	0.3 dpa

Top and Bottom Shields

Shield thickness	
Portion between vacuum vessel and coils	95 cm
Intercoil shield	<u>30 cm</u>
	125 cm

Table 6-6. (continued)

Shield composition	
First 95 cm	Graded shield consisting of alternating layers of Nitronic-33 and water, clad by 5 cm lead
Intercoil shield	Nitronic-33 plus hydrogenous material, with removable portions at finger joints
Shutdown dose rates	
Maximum at finger joint, with plug removed	20mrem/hr at 36 hours
General level in cell - with plugs in place	25 mrem/hr at 24 hours

^aAlloy type not critical.

Fig. 6-20. Finger joint shielding detail.

Fig. 6-21. Lifetime radiation doses to insulation in the outboard legs of FED-R.

The amount of intercoil shielding needed between the outboard TF coils was extrapolated from FED Baseline information. Since the cooling water in the coils is unborated, it makes little sense to borate the water in the intercoil shielding; however, lead cladding is recommended on the outside surfaces to attenuate shutdown gammas arising from the steel in the shield.

With respect to the analysis of the upper and lower bulk shields, a two-step procedure was utilized: First, a one-dimensional transport calculation of operating levels and reaction rates with the ONEDANT code, followed by a three-dimensional point-kernel calculation of shutdown gamma dose rates with the PATH code.

The ONEDANT calculations assumed the bulk shield configuration listed in Table 6-7. A wall loading of 0.5 MW/m^2 was used, so the results must be scaled up to account for the Stage II wall loading of 1.3 MW/m^2 .

Radionuclide inventories were calculated for both the last shield layer and for the TF coil. Since it is planned to attenuate shield shutdown gammas by a 5-cm lead layer on top of the bulk shield, the radionuclide inventories in the coil are of more concern; these are shown in Fig. 6-22 for the several shield compositions considered and a 0.5-MW/m^2 wall loading. The substitution of B_4C for water in the shield is detrimental from the standpoint of Cu64 activity in the coil.

The General Atomic PATH gamma shielding code was next utilized to obtain shutdown dose rates near the coil finger joint, using the geometry shown in Fig. 6-20. Source terms supplied to PATH were the Co60 and Cu64 distributions shown in Fig. 6-22 for the 50 percent Nitronic-33/50 percent borated water case; they would be little changed for non-borated water. The 12-hour Cu64 activity was decayed to 36 hours after shutdown; Co60 was calculated for five years' operation at 50 percent load factor. Results of the PATH calculations showed a dose rate of 20 mrem/hr at the bottom of the finger joint, dropping to less than 1 mrem/hr at the top. To maintain these levels with 1.3-MW/m^2 wall loading, the assumed 90-cm bulk shield would have to be increased to 95 cm. Furthermore, activated structural members and the intercoil shielding will have to be lead-clad

Table 6-7. Radii and compositions for ONEDANT poloidal model of upper bulk shield

Radius (cm)	Region	Composition
0-64	Neutron source	Void
64-160	Remainder of plasma	Void
160-165	Armor	Graphite
165-173	Vacuum vessel	10% SS 316, 90% water by volume
173-218	First shield layer	97.5% SS-316, 2.5% water
218-240	Second shield layer	92.5% Nitronic 33, 7.5% water
240-263	Third shield layer	Various ^a
263-300	Portion of TF coils	90% Cu, 10% water

Material compositions

SS-316

Fe = 0.544 at/b-cm

Cr = 0.0167

Ni = 0.0115

Mn = 0.00176

Mo = 0.00151

Nitronic-33

Fe = 0.0552

Cr = 0.0162

Ni = 0.0024

Mn = 0.011

Graphite

C = 0.083

^aThe composition of the third shield layer (23 cm thick) was varied in this study to observe the effects on activation of the TF coils, as well as the shield itself. The five compositions investigated were:

1. 25% Nitronic 33, 75% water (same as FED baseline)
2. 25% Nitronic 33, 75% borated water (0.4 at % ¹⁰B)
3. 50% Nitronic 33, 50% water
4. 50% Nitronic 33, 50% borated water
5. 25% Nitronic 33, 75% B₄C (B = 0.0825, C = 0.021)

Fig. 6-22. Saturated (a) Co-60 and (b) Cu-64 activation in upper coil.

in the vicinity of the finger joint to eliminate shutdown dose rate contributions from these sources.

6.2.4 Open Issues

There are several key shielding issues that require further investigation in the next phase of FED-R design:

1. No calculations have been made of neutral beam, vacuum-pumping, or water-coolant duct streaming. FED-R layouts indicated a possible problem where the NBI ducts pass by the TF coils. Radiation-induced insulator damage might cause shorting of adjacent plates in a coil, but the possibility of developing an arc is very small because of the low voltage applied to each coil. Furthermore, the demountable outboard coil leg could be replaced, if absolutely necessary.
2. The feasibility of thick bulk shielding on top and bottom, but thin shielding at the outboard side needs to be confirmed by 3-D Monte Carlo calculations, taking into account the shielding effectiveness of various nuclear test assemblies. An MCNP model has been developed, but only a few runs were made.
3. More detailed calculations of the shutdown dose rates at the finger joints are needed, which include contributions from shielding, structure, and bolts.
4. Maintenance procedures must be developed for the finger joints, which clearly define how contact maintenance procedures integrate with remote handling (see Sect. 9.2).

6.3 NUCLEAR TEST ASSEMBLIES

6.3.1 Location of Test Regions

In some designs for fusion test reactors, the nuclear test modules are placed inside the vacuum vessel. However, the nuclear test region in FED-R is located outside the vacuum vessel for the following reasons:

- The test assemblies will have a wide variety of shapes, sizes, and purposes, as illustrated in Sect. 6.3.2. Many of these configurations cannot physically be located within the vessel. Many practical nuclear test assemblies will not be simple cube-like or cylindrical plug modules, and it is intended that FED-R can test prototypes of such assemblies.
- Replacement of a test assembly that penetrates the vacuum vessel would necessitate bringing the torus up to air, which must be preceded by a lengthy detritiation procedure. A subsequent downtime is needed for re-establishing the vacuum.
- Leakage of a coolant line or other component of an in-vacuo test assembly could necessitate shutdown of the reactor. Such leakages are to be expected in experimental systems.
- In-vacuo test assemblies would have to be specially designed to minimize outgassing — a difficult task for each of the large variety of test systems. Each assembly would also have to incorporate a means of detritiation before removal.
- In-vacuo test assemblies must be protected by a cooled liner, which would be nearly as neutronically thick as the reference outboard first wall.
- Test modules located external to the vessel can be replaced rapidly by straight-up vertical lifting with the overhead crane.

Only the outboard region of FED-R is to be used for nuclear testing. However, there is the capability for changing the materials behind the inboard and top walls in order to vary the spectrum of scattered neutrons incident on the outboard wall.

The perimeter of the outboard region is divided into 24 test sectors, each of 1.2 m width. A test assembly up to 1.6 m wide can be removed by radial or vertical translation between TF coils. Test systems in the locations underneath the coils must be displaced toroidally before they can be removed. Three bays or six test sectors are intercepted by neutral beam injector ducts in Stage I (eight in Stage II) and one sector is occupied by plasma monitoring devices. Thus, 17 sectors are available for nuclear testing in Stage I, with a total first-wall area of approximately 60 m².

6.3.2 Test Assembly Configurations

Figures 6-23 through 6-27 show simplified plan and elevation views highlighting five different geometries for nuclear test assemblies. The basic characteristics of these configurations are described below.

- A "standard" module of 1.2-m width can be as tall as 3.0 m and have depth up to 1.3 m (by removing 0.5 m of shielding behind the nominal 0.8 m depth for testing). See Figs. 6-23 and 6-24. The total number of "standard" modules in place at any time could be as large as 34 if each is 1.5 m tall.
- Space exists for six modules of nearly 1 m³ on top of the three beam injector ducts in Stage I (Fig. 6-25). This size is the minimum that is meaningful for neutronic investigations. The space below the ducts has difficult access, but could be used to hold samples undergoing lifetime radiation tests (up to 4 MW-yr/m² exposure).
- Partial loops (up to 180° in extent) holding flowing liquid metals, flowing pebble beds, or "stringers" can extend from the top of the reactor, down past the entire length of the outboard wall, and exit at the bottom (Fig. 6-26). Loops greater than 1.6 m in width can be installed if the outboard leg of a TF coil is first withdrawn.
- Long straight fuel rod assemblies (>3 m length) can be installed parallel to the outboard wall and serviced from the top and bottom (Fig. 6-26).
- Intense neutron beams of large cross section can be "extracted" in a shielded duct extending from the first wall (Fig. 6-27).

6.3.3 Candidate Nuclear Tests

The maximum radiation levels at the outboard first wall are the following:

Fig. 6-23. Elevation view of FED-R nuclear testing facility.

Fig. 6-24. Plan view of FED-R nuclear testing facility.

Fig. 6-25. Small nuclear test assemblies installed in a neutral-beam sector in FED-R.

Fig. 6-26. Rod array and flowing-bed nuclear test assemblies having large vertical extent.

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Fig. 6-27. Illustrative neutron beam arrangement in FED-R.

	Stage I	Stage II
Fusion neutron current (n/cm ² /s)	2×10^{11}	6×10^{13}
Fast neutron flux (n/cm ² /s @ $E \geq 0.1$ MeV)	8×10^{13}	3×10^{14}
Total neutron dose rate, rad(Si)/s (all energies)	2×10^5	6×10^5
Gamma dose rate, rad(Si)/s (mean energy ~ 1 MeV)	6×10^4	2×10^5
Soft X-Ray flux (W/cm ²)	1 to 5	
Pulse length (s)	≥ 1000	≥ 1000
Integrated weekly neutron dose @ 30% duty factor	3×10^{10} rad (Si)	1×10^{11} rad (Si)

Table 6-8 lists illustrative nuclear tests and the corresponding requirements for fusion neutron production. The neutronics-related experiments can be performed satisfactorily with low neutron intensities and negligible fluences.

In the absence of confirmatory measurements, the neutronic performance of an assumed distribution of blanket modules in a magnetic confinement fusion (MCF) reactor cannot be known with assurance because of the uncertainties in coupling of the geometrically extended fusion neutron source with the complicated neutron scattering and absorbing assembly established by the reactor components. Uncertainties in this distribution will contribute to the overall uncertainty in calculating such quantities as the global tritium breeding or fissile breeding ratio of a full-scale reactor, as well as the performance of single, isolated blanket modules in industrial fusion facilities.

Point-neutron source experiments cannot establish the neutron energy distribution that results from the scattering phenomena characteristic of toroidal fusion reactor phenomena, but only give reaction cross sections and breeding data in ideal assemblies with effectively perfect blanket coverage.

Most of the heating-related experiments such as thermal hydraulic and thermomechanical testing require Stage I fluxes but relatively low fluences. Investigations of materials compatibility (breeder, clad, coolant, structure) require higher fluences. Thermal cycling phenomena that involve materials compatibility as well as cracking of sensitive

Table 6-8. Illustrative nuclear tests

	14-MeV neutron requirement	
	Source Intensity (n/s) ^a	Fluence Per Run (n/cm ²) ^a
Neutronics-related experiments		
• Activation and afterheat	10 ¹¹	10 ¹³
• Shield configuration and duct streaming	10 ¹³	10 ¹⁵
• Tritium breeding profiles in single, limited coverage modules	10 ¹²	10 ¹⁵
• Flux modification by variation of reactor albedo	10 ¹²	10 ¹⁶
• Flux enhancement in test modules by neutron multipliers	10 ¹²	10 ¹⁵
• Neutron beam development	10 ¹²	10 ¹⁶
Heating-related experiments		
• Energy deposition profiles	10 ¹³	10 ¹⁶
• Power enhancement by multipliers	10 ¹²	10 ¹⁶
• Transient temperature responses	10 ¹³	10 ¹⁶ -10 ¹⁷
• Thermal cycling and materials compatibility		
• Electrical power demonstration	10 ¹³	10 ¹⁶ -10 ¹⁷
• Tritium recovery	10 ¹³	10 ¹⁸ -10 ¹⁹
Radiation damage testing		
• Transient electronics effects	10 ¹³ -10 ¹⁵	
• Permanent effects on hardened electronics	10 ¹²	10 ¹⁶ -10 ¹⁷
• Organic insulators	10 ¹²	3 × 10 ¹⁸
• Inorganics, including solid breeders	10 ¹³	3 × 10 ²⁰
MHD-related studies		
• Liquid metal flow	N/A	N/A
• Tritium migration in liquid metal	10 ¹³	10 ¹⁷

^a14-MeV neutron flux at vessel wall.

components can be studied with 10,000 or more nuclear pulses of relatively short duration (e.g., 100 s). Short-pulse blanket module tests would also yield engineering data needed for checking calculational models of system transients caused by power pulsing. During this "pulsed" operation, the magnetic fields, injected beam power, and plasma parameters would actually be unchanged (to minimize fatigue of reactor components), but all tritium fueling would be replaced by deuterium fueling to create periods of deadtime for neutron production.

Power and flux enhancement

For some nuclear heating-related tests, it is desirable to have a larger blanket power position than results from a wall loading of 0.4 MW/m². For other tests such as radiation damage, it is desirable to increase the neutron flux in a test module. Power density and/or neutron flux could be enhanced in a test module by the strategic placement of uranium in or around the test module region. (See Fig. 6-24.) Practical considerations of cost and access to fissile material dictate the use of depleted or natural uranium, with careful positioning of ⁶Li or ¹⁰B to absorb slow neutrons and thereby minimize capture in ²³⁸U. The advantages of this technique over the proposed use of fission reactors for testing of fusion blanket components are the following:

- Much larger test volume is available in a large tokamak device (e.g., 1,000 liters per module) than in a fission reactor irradiation facility.
- By removing the uranium inserts from the tokamak module, reactor-relevant profiles of flux spectra and energy deposition can be determined.
- The spatial gradients of power density and fast-neutron flux can be controlled, in accordance with simulation requirements.

Electrical power production

Six test sectors, or 35 percent of the total available, have a first wall area of 20 m² exposed to a total of 8 MW of fusion neutrons in Stage I. About 3 MW of electrical power could be generated by the

nuclear heat deposited in these sectors, and this level can be raised to at least 15 MW if the blanket assemblies were to contain depleted uranium to the extent of approximately 10 percent by volume. The uranium can be inserted by means of vertical rods or by stringers in loops.

Radiation damage testing

Test modules can be used for nuclear radiation damage evaluation and hardening programs for electronic devices, circuits, fibers, and common organics. The relatively low neutron flux, but large fluence over the course of several months, imply that Stage I operation of FED-R would, in general, be suited best for determining the permanent damage of neutron and gamma radiation to dose-dependent devices and systems.

A very large number of samples can be exposed simultaneously to a nearly uniform radiation field. This opportunity would allow determination of the minimum dose that results in damage, by removing the samples one at a time after each of a series of fusion pulses, for mechanical and electrical tests. Transient incapacitation effects on electronic devices and crystals that are sensitive to fast-neutron fluxes of 10^{13} n/cm²/s or lower can be evaluated by in situ monitoring.

As suggested in Fig. 6-24, the neutron spectrum incident on the test samples can be adjusted by suitable choice of materials placed inboard of the reacting plasma, and by changing the size and composition of reactor components near the test modules. For example, by surrounding a test assembly with beryllium or lead of various thicknesses, the neutron fluxes in the assembly can be increased significantly and the exposure symmetry modified. The gamma-ray field in a tokamak reactor originates from fast neutrons striking reactor components, and can be independently tailored in spectrum and enhanced or suppressed in intensity by appropriate choice of materials and components in the test module sectors. Because of the extended source geometry and extended irradiation test volumes, it is possible to establish different neutron spectra and different ratios of neutron to gamma dose rates simultaneously in different test modules, as well as in neutron beams extracted for external use.

Neutron beams

FED-R can produce fast neutron beams of much larger cross section and higher intensity than available by any other technology at these energies (<15 MeV). In principle, polarized neutron beams can be generated by spin-polarizing the reactant deuterons or tritons. Application of these large-area neutron beams is discussed elsewhere.

6.3.4 Servicing of Test Assemblies

Various machines such as coolant connect/disconnect devices, fuel rod changers, sample retrievers, and module positioners are required to service the nuclear test systems. These machines would not be positioned near the reactor during operation in order that their activation levels can be minimized. The machines would be kept behind shielding in the reactor cell during FED-R operation.

The position of the TF coil joints was selected to allow straight-up vertical removal or installation of oversized test modules. The advantages of pulling test modules out vertically rather than radially are the following:

- A relatively easy crane motion, compared with a more complicated manipulator system required for radial removal.
- No interference with large equipment on the reactor room floor.
- Test modules can be taken directly to the hot cell by the crane.

6.4 ELECTROMAGNETIC ANALYSIS

The FED-R has an attractive feature of providing good conducting surface near the plasma for reducing the thermal energy on the surface of the vessel due to a disruption. This reduces the possibility of melting of the surface, and can help reduce the number of disruptions. By providing a good surface for large eddy currents, the plasma is held away from the walls for a longer time during a disruption. This allows the magnetic stored energy in the plasma loop to transfer to I^2R heating in the structures or to allow the feedback control systems to produce

changes in the control fields to eliminate the current quench phase of the disruption. If a good passive conductor is not provided, the plasma moves towards the wall at a much faster rate, and it can hit the walls before its magnetic energy can be absorbed by the I^2R heating of the structures. The magnetic energy is transformed into plasma heating. The heat loads are, therefore, much higher at the locations where plasma touches the walls.

The plasma position control system is relatively inexpensive for FED-R because the good conducting eddy current path or structure near the plasma provides a self-stabilization of the plasma due to minor disturbances. The control fields can be located near the plasma but need not be inside the vacuum chamber. All these features result in smaller position control coils and control systems.

The FED-R torus design has a relatively short length for radial current flow, which occurs at the torus joints. This results in reduced forces and torques on the structure.

Because there are no segment gaps or spaces between segments, there will be no arc damage during a disruption; and sufficient contact material can be provided at the torus joints to conduct the several million amperes of induced eddy currents without welding the surfaces together.

The torus configuration for the FED Baseline designs is selected on the basis of mechanical considerations. These include the desire to have the vacuum surface behind the neutron shield and to allow partial hands-on maintenance of the vacuum seals or welds. However, FED-R torus cutting and welding will be a fully remote operation. The radiation may result in a permanent change in the metal which could preclude rewelding. To overcome this problem, vacuum vessel seal welds are protected by a localized shielding.

The electromagnetic effects of torus structure for existing machines are reduced because the vacuum vessel conductance is decreased with insulating breaks, bellows or high resistance structures. This approach is impractical for reactor relevant designs. The torus design must change with increases in the burn time, neutron loading, cooling, disruption energies, and forces. The structure design must de-emphasize the reduction in the eddy currents in structures and must maximize the

eddy currents in the torus structure adjacent to the plasma and minimize the eddy currents in the remaining structures which are closely coupled to the control and startup windings.

The FED-R startup utilizes rf assist provided by ECRH. Production of the initial plasma by ECRH results in a reduction of the voltage applied across the plasma loop from a hundred or more volts to approximately 15 volts. The startup coil size and power required are low, and the energy loss in the structure is reduced by about two orders of magnitude with rf assist startup. This results in a reduction of startup energy and could lead to elimination of electrical breaks in the supporting structures and components attached to the torus, such as the coolant pipes and diagnostics.

The present specifications for the torus time constant is 100 milliseconds. The FED-R basic design lends itself to changing the resistivity to accommodate detailed designs for decay control of disruption currents.

REFERENCES

1. *Fusion Engineering Device*, DOE/TIC-11600 (October 1981).
2. C. A. Flanagan, D. Steiner, and G. E. Smith, *Fusion Engineering Device Design Description*, ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee, December (1981).
3. D. L. Chapin and W. G. Price, "Neutron Wall Loading Distributions in Tokamaks," *Trans. Am. Nucl. Soc.* 21 (1975), pp. 66-67.
4. E. T. Cheung and B. A. Engholm, *Trans. Am. Nucl. Soc.* 38 (1981), 217-8.
5. B. J. Micklich, D. L. Jassby and C. E. Clifford, *Trans. Am. Nucl. Soc.* 41 (1982), 172-3.
6. L. P. Ku, "Nuclear Radiation Analysis for TFTR," Princeton Plasma Physics Report PPPL-1711 (Sept. 1980).
7. D. L. Jassby, "Potential of the TFTR as a Fusion Test Facility," in *Proc. Conf. on Fast, Thermal and Fusion Reactor Experiments* (ANS, Salt Lake City, 1982), Vol. II, pp. 293-304.
8. D. L. Jassby, "Tokamak Fusion Generators for Nuclear Radiation Effects Testing," in *Proc. 1982 Annual Conf. on Nuclear and Space Radiation Effects* (IEEE, Las Vegas, July 1982) to be published in *IEEE Trans. Nucl. Sci.*

7. TRITIUM SYSTEMS

The FED-R tritium systems have three primary purposes: to provide tritium and deuterium fuel for the reactor; to provide tritium handling that limits worker radiation exposure; and to minimize tritium losses to the environment.

7.1 DESIGN REQUIREMENTS

The most important design requirement for the tritium system is the throughput of the primary fuel cycle (Table 7-1). This leads to a steady-state DT flux of 0.11 standard temperature and pressure (STP) liters s^{-1} for continuous Stage II operation, assuming a 3 percent burnup fraction. The amount of tritium consumed is proportional to the fusion power level and the capacity factor. Values of 1.3 kg/yr for Stage I operation and 4.3 kg/yr for Stage II operation were calculated assuming 32 percent reactor capacity factor.

The composition of the reactor exhaust is expected to be 48.5% T_2 , 48.5% D_2 , 0.5% N_2 , 1% DC_4 or CT_4 , and 1% He. Other possible compounds include T_2O , D_2O , ND_3 , NT_3 , and their hydrogen isotopic mixes. Deuterium and tritium are required to be returned to the reactor 99+% isotopically pure with the balance being the other species (i.e., 99% D, 99% D/1% T). The tritium system will have a tritium inventory of ~ 0.7 kg for continuous Stage I operation and ~ 1.5 kg for continuous Stage II operation. This is based on the need to maintain a 30-day burn inventory.

Release of tritium to the environment, both operating and accidental, will be minimized and will in no case exceed the existing guidelines and regulations. Tritium levels in all buildings will be maintained below 500 $\mu Ci/m^3$, except during a postulated accidental release. The release of tritium from the plant from all sources will be < 20 Ci/day averaged over routine operation, maintenance, and in-plant accidental releases.

7.2 FUELING SYSTEM

The fueling system requirements are based on the Stage I and Stage II data given in Table 7-2. Because a large part of the fueling

Table 7-1. Reference FED-R tritium parameters

Parameter	Stage I	Stage II
Fusion power, MW(th)	75	250
Burn time, s	10,000	28,000
Cycle time, s	10,250	28,250
Tritium fractional burn up, %	2.0	3.0
Particle (atom) flux pumped, s^{-1}	2.7×10^{21}	6.0×10^{21}
Steady-state D-T flux, l/s (STP)		
Consecutive pulses	0.050	0.111
Tritium burned, g/day		
Consecutive pulses	11.5	38.4
Tritium process inventory, g		
(Does not include first wall inventory)		
Storage	350	1150
Vacuum pumps (turbo)	$<10^{-3}$	$<10^{-3}$
Pellet fueler	25	25
Fuel purification	250	300
Miscellaneous	50	50
Total	675	1525
Tritium consumed, kg/year	1.33	4.33
(32% capacity factor)		

Table 7-2. Data base for FED-R fueling system

Parameter	Stage I	Stage II	Units
Average fusion power	75	250	MW
Cycle time	10,250	28,250	s
Burn time	10,000	28,000	s
Tritium burn rate	2.68	8.93	$10^{19}/s$
Burn fraction	2.0	3.0	%
NB fueling rate	2.1	2.5	$10^{21}/s$
Pellet fueling rate	1.2	4.9	$10^{21}/s$
Gas puffing rate	0.60	2.00	$10^{22}/s$
Plasma density, N_o $N/N_o = 1 - (r/a)^2$	0.50	1.70	$10^{20}/m^{-3}$
Electron temperature (T_o) $T/T_o = [1 - (r/a)^2]^2$	13.5	18.0	keV
Plasma dimensions			
-Major radius	3.50	3.50	m
-Minor radius	0.85	0.85	m
-Elongation	1.50	1.50	
Plasma inventory	.008	.0264	g

is supplied with neutral beam injectors, small pellets are needed to maintain a reasonable pellet injector rate ($\geq 4/s$). The density and temperature distribution given in Table 7-2 and shown graphically in Fig. 7-1 prevent the pellets from penetrating deeply into the plasma.

Figure 7-2 shows the pellet velocity needed to penetrate a fraction of the plasma volume and mass for 2, 3, and 4-mm pellets. The three curves drawn for the 3-mm pellet illustrate the effect of a ± 10 percent mass variation in pellet production. For a 3 mm pellet having a velocity of 2 km/s, the plasma mass fraction fueled directly is only 35 percent although the corresponding volume fraction is about 58 percent. For a 2-mm pellet, the corresponding values are 29 percent and 54 percent, respectively. Because of the rapid rise in electron temperature with plasma depth, the 2-mm pellet is nearly as effective as a 3-mm pellet and there is little to be gained by increasing the pellet velocity. Considering these limitations, it is best to fuel the core of the plasma with tritium and deuterium neutral beam injectors and the outer part of the plasma with 2-mm frozen pellet injectors.

Based on the above considerations, the fueling system requirements are given in Table 7-3. Note that a 2-mm pellet provides an injection rate of 4 s^{-1} during Stage I and 14 per s^{-1} during Stage II. Because of the lower pellet rate during Stage I, one pellet injector should provide satisfactory availability. The major design data for the pellet injectors and gas puffers are given in Table 7-4.

The fueling is implemented with gas puffers, frozen pellet injectors, and NB injectors. Only the first two are described here, the latter is discussed under plasma heating. Prior to startup, the torus is highly evacuated and backfilled with deuterium gas puffers. Gas puffing continues until the plasma temperature is about 1 keV. Frozen D and T pellet injection and NB injection are then coordinated to control the plasma density and temperature.

Pellet fuel injectors are located as shown in Fig. 7-3. The water-cooling header transition section has adequate shielding for Stage I operation but may require an add on cylindrical sleeve shield for Stage II operation. Each injector can inject 2-mm diameter D and T

Fig. 7-1. FED-R normalized plasma density and electron temperature distributions.

Fig. 7-2. FED-R pellet velocity/normalized plasma cumulative volume and mass.

Table 7-5. Pellet injector requirements for FED-R

Parameter	Stage I	Stage II	Units
Number of dual channel pellet injectors	1	2	
Pellet size: tritium	2.0	2.0	mg
deuterium	1.3	1.3	mg
Pellet diameter	2	2	mm
Injector distance to plasma	5	5	m
Scatter cone half angle	0.5°	0.5°	
Magnetic field in vicinity of pellet injector	0.4	0.5	T
Pellet velocity	≥ 2000	≥ 2000	m/s
Pellet rate/channel	2	7	s ⁻¹

Table 7-4 . Pellet injector design data

Parameter	Stage I	Stage II	Units
Number of injectors	1	2	
Pellet size (d = h)	2	2	mm
Pellet velocity	≥ 2000	≥ 2000	m/s
Maximum injection rate per channel	2	7	s ⁻¹
Injection rate range	0-5	0-15	s ⁻¹
Acceleration rate	5	15	s ⁻²
Injection period	10,000	28,000	s
Purge period	250	250	s
Scatter cone half angle with collimator tube	0.5°	0.5°	
Maximum tritium inventory per injector	25	25	g
Number of deuterium gas puffing systems	2	2	
Number of gas puffing nozzles (1 per torus sector)	12	12	
Maximum gas flow rate	0.5	0.5	g/s
Maximum valve pulse length	200	200	ms
Maximum valve pulse rate	5	5	s ⁻¹

Fig. 7-3. Elevation section of the FED-R showing the pellet injector and interfacing assemblies.

pellets concurrently. A mixed specie fuel may be substituted for the D or T as shown in Fig. 7-4. Downstream pressure regulators control the flow of gas to the nitrogen cooled chiller tanks located external to the pellet injector. Remote controlled selector valves determine the fuel gas admitted to the liquefiers. The liquefiers are located inside the insulated fuel injectors. The liquid fuel to the pellet former flows by gravity feed. The pellets are then admitted to the dual feed pellet injector at a rate controlled by a process computer. The programmable computer uses electron density and neutron measurement to calculate the pellet injection rate of both channels. Either fuel injector will meet the requirements, but the pellet rate can be doubled by operating both injectors concurrently.

7.3 FUEL CYCLE AND TRITIUM SYSTEM

The FED-R fuel cycle and tritium system, illustrated in Fig. 7-5 is designed for continuous Stage II operation. The tritium systems can be logically subdivided into four categories: Primary Fuel Cycle, Secondary Systems, Building Detritiation Systems, and Tritium Systems Data Acquisition. These systems are described in Ref. 1 and are summarized below.

7.3.1 Primary Fuel Cycle

The primary tritium fuel cycle takes the reactor exhaust from the vacuum pumping system and routes it to the Fuel Clean-up unit (FCU). The FCU removes all impurities except helium to below 1 ppm total, so that it becomes suitable feed for the Isotope Separation System (ISS). The ISS isotopically separates the deuterium and tritium by cryogenic distillation. A third stream from the ISS containing protium, some deuterium, and helium is released to the stack if it contains a sufficiently low amount of tritium or, if necessary, to the tritiated water recovery unit for further tritium stripping. The deuterium and tritium that is 99% isotopically pure and 99.99% elementally pure is stored as a metal hydride before being recycled to the cryogenic pellet fuelers.

Fig. 7-4. Proposed pellet fueling system showing one of two injectors.

Fig. 7-5. FED tritium systems.

New tritium to replace that burned in the reactor is brought into the tritium processing building in approved shipping containers and unloaded in the tritium receiving and storage system. This system also supplies limited storage for the new tritium awaiting processing in the ISS. Also considered part of the primary fuel cycle is the tritium analysis system, which will have the ability to analyze all important tritium streams (both isotopically and elementally) to support both process control and accountability.

7.3.2 Secondary Systems

The FED-R will incorporate secondary and tertiary tritium containment to minimize worker exposure and to limit environmental tritium releases, as well as to minimize the tritium loss from the primary fuel loop. All primary fuel cycle components will incorporate secondary containment; higher risk components will use tertiary containment.

The tritium waste treatment (TWT) system takes the impurity load from the FCU and removes the last traces of tritium before releasing the impurities to the stack. This system uses hydrogen oxidation/water adsorption to effectively remove the tritium and is similar to systems used in TSTA and TFTR. The glovebox detritiation system (GDS) serves to detritiate glovebox and secondary enclosure atmospheres, whether in a recycle mode or in a purge mode. Both the TWT and GDS systems will produce tritiated water as a byproduct, but the TWT water will have a much higher tritium concentration. The GDS will also be used to ventilate temporary maintenance enclosures within the tritium processing building to further reduce tritium effluents. Because the TWT and GDS are similar, they provide a redundant capability.

Large quantities of tritiated water may be produced by the building detritiation systems and the reactor first wall coolant loop as well as by the TWT and GDS. All of this tritiated water will be processed in the tritiated water recovery (TWR) system. This system utilizes the combined electrolysis catalytic exchange technology and cryogenic distillation to recover all the tritium and return it to the primary fuel cycle.

The final secondary system is the tritium waste processing (TWP) system. This system will provide the capability to package all tritiated wastes in containers suitable for disposal at commercial sites.

7.3.3 Building Detritiation Systems

The building detritiation systems provide primary tritium removal from the atmosphere in the hot cell facility, secondary containment in the reactor building and tertiary containment in the tritium processing building. In addition, an added emergency detritiation capacity is available for the reactor building.

The hot cell detritiation (HCD) system is sized to handle a continuous tritium release of 100 Ci/day while maintaining the hot cell atmosphere at $500 \mu\text{Ci}/\text{m}^3$. The primary purpose of the reactor building detritiation (RBD) system is to remove tritium from the reactor building. In addition to handling tritium permeating through the torus vacuum boundary and leaking from coolant systems during normal operation, the RBD also provides for component offgassing during maintenance of the torus. This system is sized to handle a continuous tritium release of 100 Ci/day while maintaining the reactor building atmosphere at $500 \mu\text{Ci}/\text{m}^3$ and at a pressure negative to the outside. The emergency detritiation system (EDS) is similar to the HCD and RBD but has twice the capacity of either. The EDS would operate only if a large release of tritium were to occur in the reactor building or in the tritium processing building. For a single release of 25 g (pellet fueler inventory) in the reactor building, resulting in an initial tritium concentration of $2.2 \text{ Ci}/\text{m}^3$, the EDS, along with the RBD, could reduce the tritium concentration to $5 \mu\text{Ci}/\text{m}^3$ in about 60 hours. For a more typical release of 1 g in the reactor building, 45 hours would be necessary to reduce the level to $5 \mu\text{Ci}/\text{m}^3$.

This system accurately measures performance parameters and provides control of components, subassemblies, and the tritium systems as a whole; it assures adequate safeguarding and accountability for the large tritium inventory, and it provides alarms for "out of limit" processes.

REFERENCES

1. *Fusion Engineering Device*, DOE/TIC-11600, October (1981).
2. C. A. Flanagan, D. Steiner, and G. E. Smith, *Fusion Engineering Device Design Description*, ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee (December 1981).

8. FACILITIES DESCRIPTION

The essential purpose of the plant facilities is to support the operation and maintenance of FED-R. These facilities provide housing and a controlled environment for the FED-R equipment. The types and sizes of the buildings and structures are expected to be very similar to the facilities described for the FED baseline design.¹ The few differences account for different design concepts and service requirements. The suggested site arrangement and the general building floor plans of the FED baseline design¹ are also applicable to FED-R. The major differences are in the sizes of the required facilities. The estimated sizes for the FED-R facilities are summarized in Table 8-1.

The FED-R facilities have not been studied in detail, and for the most part the FED baseline results have been accepted without revision. Only those facilities are changed where differences in device design influence the facilities description, with the exception of the tritium processing building. The tritium processing building is changed to reflect more detailed equipment layout and sizing studies.

8.1 REACTOR BUILDING

The size of the reactor building is changed to accommodate the different size of the FED-R device. The use of neutral beams in place of rf heating has increased the needed floor area for the FED-R. However, a reduction in overall device height and the height of the tallest component to be lifted out of the center of the device has resulted in a smaller building height. The combination of these factors results in about 10 percent smaller building volume for FED-R, compared with that for the FED baseline.

No attempt is made at this point in the design to investigate the impact of the differences in heat removal systems for the FED-R. The first wall helium coolant system and the TF coil water cooling system together require substantially more floor area for heat exchangers and other cooling system equipment. The current approach places this equipment in the reactor building, but no sizing or equipment layout

Table 8-1. FED-R structures and site facilities

Description	Outside Dimensions l x w x h (m)	Approx. Volume (m ³)	Approx. Floor Area (m ²)
● Reactor Building and Hot Cell Facility	-		
Reactor building	64×57×36*	131,000	3670
Hot Cell Facility	78×52×32	130,000	4060
● Cooling System Structures	-		
Intake Structures	15×15×6	1,350	225
Discharge Structures	-		
Intake and Discharge Conduits	-		
Recirculating Structures	45×15×10	6,750	675
Cooling Tower Systems	-		
Cooling Towers	-		
Recirculating Water System	-		
● Power Supply and Energy Storage Bldgs.			
Copper and TF Coil Electrical Eqpt. Bldg.	100×56×10*	56,000	5600
OH and PF Electrical Equipment Bldg.	88×43×10	37,840	3784
Bulk Heating Electrical Equipment Bldg.	26×16×10	4,160	416
Motor Generator Flywheel Bldg.	80×20×20	32,000	1600
● Miscellaneous Buildings	-		
Auxiliary Buildings	-		
Cryogenic Refrigeration Building	20×15×10*	3,000	300
Tritium Processing Building	50×30×20*	30,000	1500
Ventilation Building	58×25×30	43,500	1450
Radioactive Waste Building	40×40×10	16,000	1600
Control Room Building	(30×15×10)**	(4,500)**	(450)**

* Changed from FED Baseline, ORNL/TM-7948, Chapter 9

** Included in Administration Building

Table 8-1. FED-R structures and site facilities (Cont)

Description	Outside Dimensions l × w × h (m)	Approx. Volume (m ³)	Approx. Floor Area (m ²)
Diesel Generator Building	32×16×10	5,120	512
Adminstration Building	90×60×10	54,000	5400
Service Building	-		
Mockup and Shop Building	65×65×40*	169,000	4225
Miscellaneous Structures and Bldg. Work	-		
Control Room Tunnel			
• Ventilation Stack	10 m Dia × 100		

studies have been performed to estimate this potential impact on reactor building size.

8.2 HOT-CELL FACILITY

The FED baseline hot cell facility is currently being revised, but the studies leading to design revision have not been completed. When the revisions are incorporated into the baseline design, they are also expected to be applicable to the FED-R design. Until the baseline is revised, the hot cell facility for FED-R is shown to be the same as the facility described in Ref. 1. More detailed investigation of the hot cell for FED-R facility would probably show a need for some increase in size over the baseline to accommodate neutral beam maintenance and testing.

8.3 ADDITIONAL FACILITIES

TF Coil Electrical Equipment Building

The TF coil electrical equipment building volume is expanded by more than 160 percent to accommodate the larger power supply requirement of the copper TF coils. Additional space is also required near the building for the large number of transformers which are mounted outside the building. This large increase in the TF electrical equipment building also impacts the general building arrangement.

Cryogenic Refrigeration Building

The need for cryogenic refrigeration is greatly reduced for FED-R. Thus, this building size has been reduced by 75 percent from the FED baseline.

Tritium Processing Building

The changes in the tritium processing building reflect increased detail in the sizing of tritium processing equipment and equipment

arrangement. The building volume of the FED-R design is 57 percent of the volume for the same building for the FED Baseline.¹

Mockup and Shop Building

This building size has been reduced by 14 percent to allow for the smaller FED-R device dimensions.

8.4 ENVIRONMENT AND SAFETY

The environmental and safety aspects of the FED-R design are expected to be very similar to the FED baseline. The types and quantities of hazardous material remain the same. Some second order environmental impacts may change when the emphasis is placed on copper procurement instead of superconductor procurement.

The potential cryogenic release accident is greatly reduced for FED-R. However, the water in the TF magnet cooling system may lead to different accident scenarios than present for the FED baseline design.

REFERENCES

1. *Fusion Engineering Device*, DOE/TIC-11600, October 1981.
2. C. A. Flanagan, D. Steiner, and G. E. Smith, *Fusion Engineering Device Design Description*, ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee, December 1981.

9. MAINTENANCE CONCEPT

Two major objectives were considered in the development of a maintainable FED-R configuration:

- provide straightforward access to all major components.
- establish a feasible replacement scenario with an acceptable downtime for all major components.

The replacement of the outer TF coil leg is the key maintenance issue. This task is studied to a first-order level of detail. Replacement of the internal PF coils and of the vacuum vessel sectors is also considered in this section.

9.1 MAINTENANCE APPROACH

Tokamak design studies since the mid-seventies have emphasized the capability to do hands-on maintenance even in a limited capacity. This design approach applies equally to this device. Personnel access around the device after a reasonable shutdown period is permitted within the limits and requirements discussed below.

The maintenance approach is three-fold and considers the requirements for contact and remote maintenance, component disassembly, and the technology available for remote handling equipment.

1. Contact and Remote Maintenance

Generally, all areas outside the device shield are capable of being maintained by contact operations beginning 36 hours after shutdown. This assumes the plasma chamber is unopened and all vessel penetrations are properly shielded. In addition, these "contact tasks" also require the capability to be totally remotely accomplished for emergency situations in which personnel entry into the reactor building would be prohibited.

2. Component Disassembly

Wherever practical, subsystems and components are designed for replacement as individual modules and for minimal impact on adjacent

systems. This is particularly important for components which are expected to require more than one replacement during the device lifetime. Examples of these are the blanket modules, divertor/pumping modules, and possibly the vacuum vessel. Access and simplicity of handling are considered configuration requirements.

3. Remote Maintenance Equipment Technology

Because FED-R is a near-term device, component maintenance and replacement assume the use of existing remote maintenance equipment and technology in the areas of manipulator systems, viewing systems, and transport systems. This requires that the configuration be developed around the capabilities and limitations of commercially available technology (although customized variations would have to be ordered).

Permissible dose rates

The dose rate was chosen to be 5 millirems per hour at outside surfaces of the device and 20 millirems per hour near an exposed TF coil joint 36 hours after shutdown. At these levels, personnel access to the reactor for maintenance operations may be limited in time, but nevertheless possible. For work around the shielded device, 5 mr/h means that a worker can accumulate up to 48 hours per week and not exceed the permissible quarterly limit for whole body exposure, which is 3 rems.

$$50 \text{ weeks} : 4 = 12.5 \text{ weeks}$$

$$3000 \text{ millirems} : 12.5 = 240 \text{ mr/week}$$

$$240 \text{ mr/week} : 5 \text{ mr/h} = 48 \text{ h/week}$$

A similar calculation near the coil joint (at 20 mr/h) indicates that 12 hours of operations would be permissible for each worker per week. These estimates of the time permitted for maintenance operations do not appear restrictive. However, it should be pointed out that if the general requirements of ALAP (as low as practicable) are applied (reference ERDA document 0524, Part VI, Section B), these estimates of allowable dose could be reduced by as much as a factor of 5.

Shutdown time

The shield thickness is designed to permit personnel access to the device outer surface 36 hours after shutdown. This waiting period is sufficient to reduce the surface dose rate to 5 mr/n and can also include general maintenance preparations such as de-energizing coils, draining coolant, and preparing maintenance equipment.

Contact maintenance boundary

Contact maintenance is limited to operations outside the device shield boundary, provided the shield is intact and major penetrations are shielded. Furthermore, contact maintenance is generally defined to mean unsuited operations. Therefore, disassembly procedures in the reactor building will include provisions for decontamination of tritium and activated particulate matter. It is a design goal for FED-R that contact operations be unsuited; however, suited operations using portable life support systems will be an acceptable mode of "hands-on" operations if contamination levels exceed maximum permissible concentrations.

Device operation

Maintenance activities are not permitted in the reactor building when FED-R is operating. This requirement results from safety considerations for personnel and the avoidance of neutron-induced activation to the maintenance equipment. In addition, general maintenance activities next to the device are prohibited when the coils are energized, even if plasmas are not being generated. A safe boundary around the device for both personnel and equipment will be determined by stray field plots in the reactor building.

9.2 REPLACEMENT OF THE OUTER TF COIL LEG

Replacement of the outer TF coil leg, characterized in Table 9-1, was estimated to take 350 hours of maintenance operations. Of this, 140 hours is required for the removal phase and 210 hours for the replacement

(replacement was assumed to require 50% more time than removal to account for the difficulties of joint alignment for both the fingers and the bolts). More than one crew can work in parallel if several coil legs are required to be removed. The maximum number of crews may be limited by the availability of maintenance equipment, workmen, and space required for the workmen and their equipment.

Table 9-1. Characteristics of the coil outer leg

• Overall size	1.1 x 1.7 x 10.6 meters high
• Weight	235 tonnes
• Joints	~50 fingers each
• Number of plates	17, each with 4 bolts
• Translation	1.5 m radially outward, then vertical lift
• Mode of operation	Contact maintenance
• Upper and lower superconducting PF coils may be left at cryogenic temperature	

There are 13 major disassembly operations required to remove the outboard leg of a TF coil. They are listed below to correspond with Fig. 9-1.

1. Remove the pellet injector.
2. Disassemble the upper PF coil coolant and electrical connections and the support columns from the lower coil (12 places).
3. Remove the upper coil with its attached supports to a laydown area.
4. Install jackstands under the lower PF coil.
5. Disconnect and remove vacuum ducts (12 places).
6. Unbolt the lower PF coil support columns at the floor (12 places).
7. Lower the PF coil clear of the TF coil.
8. Disassemble and remove intercoil structure.
9. Disconnect TF coil coolant manifold and electrical connections.
10. Disassemble the TF coil joints (2 places).
11. Install the lift fitting and attach the overhead crane hook.

Fig. 9-1. Disassembly sequence of the outer TF coil leg.

12. Install the upper and lower jackscrew fittings, remove bolts, and separate the joints.
13. Translate coil leg radially outward and up.

Caveats

The three caveats listed below apply to the operations shown above:

- a. The following are assumed to be parallel operations:
  - 1 and 2
  - 3, 4, and 5
  - 8 and 9
- b. This scenario assumes a coil leg which is not next to a neutral beam injector.
- c. Time for equipment set up was not estimated.

The following list of basic maintenance equipment is required for this scenario:

- Overhead crane.
- Upper PF coil lift fixture.
- Two TF coil joint separators (motor driven).
- TF coil lift fitting.
- Two TF coil bolt extractors (1.7 meter pull).

9.3 REPLACEMENT OF INTERNAL PF COILS

There are four resistive coils which are located within the bore of the TF coils. Three are positioned below the plasma chamber, and one is above. This subsection is a brief discussion of the disassembly steps required to replace these coils. Its purpose is to investigate the configuration for areas which should be modified to improve the maintainability of this coil system. Table 9-2 outlines the major steps which are necessary to remove the lower coils.

Table 9-2. Sequence of steps to remove the lower internal PF coils

-
1. Remove the upper superconducting coil with its stanchion supports attached.
  2. Retract the lower superconducting coil to a position below the TF coil leg.
  3. Remove five sets of intercoil support structure between four adjacent TF coils.
  4. Demount and remove four adjacent outboard TF coil legs.
  5. Remove the shield modules in front of 180° of the internal coil, below the plasma chamber support platform.
  6. Remove three adjacent pump systems; remove the shielded ducts which interface between these pumps and the chamber support platform. (This requires that the lower superconducting coil be lowered flush to the floor.)
  7. Demount the internal PF coil joints, which are 180° apart.
  8. Extract the half segment of this coil.
  9. Rotate the remaining half segment and extract it.
-

The replacement of spare coil segments (or the repaired coil) is accomplished in the reverse order of that shown above and these operations will be more difficult and time-consuming for several reasons.

- a. Alignment of components for reassembly is a necessity for welding and installing structural attachments.
- b. Assembling coil joints requires installing insulation, torquing bolts, brazing coolant lines, and testing for conductivity.
- c. In general, the reassembly of most components requires intermediate testing for electrical, vacuum, coolant, and structural integrity.

Removal of the upper internal coil is somewhat simpler because it does not require removing any pumps or their associated ducting. It also only requires removing three TF coil legs. Table 9-3 outlines the major steps necessary to remove this coil.

As an example of modification that may improve maintainability, increasing the number of joints in the PF coils (now only two) may permit a reduction in the number of components that must be disassembled before removal of a PF coil.

Table 9-3. Sequence of steps to remove the upper internal PF coil

-
1. Remove the upper superconducting coil with its stanchion supports attached.
  2. Retract the lower superconducting coil to a position below the TF coil leg.
  3. Remove four sets of intercoil structure between three adjacent TF coils.
  4. Demount and remove three adjacent outboard TF coil legs.
  5. Remove the shield modules in front of 180° of the internal coil, above and in front of the plasma chamber.
  6. Demount the internal PF coil joints which are 180° apart.
  7. Extract the half segment of this coil.
  8. Rotate the remaining half segment and extract it.
-

As an example of a modification that may improve maintainability, increasing the number of joints in the PF coils (now only two) may permit a reduction in the number of components that must be disassembled before removal of a PF coil.

9.4 REPLACEMENT OF THE VACUUM VESSEL SECTORS

The vacuum vessel is made up of six 60° sectors. Removal of three of the sectors requires moving neutral beam injectors; removal of one of these three sectors also requires disassembling the pellet injector. Otherwise, disassembly of components is identical for all of the vessel sectors. Table 9-4 outlines the major steps necessary to remove the sector that contains the pellet injector.

Table 9-4. Sequence of steps to remove a vacuum vessel sector

-
1. Remove the pellet injector (one sector only).
  2. Remove the neutral beam injector (three sectors only).
  3. Remove the upper superconducting coil with its stanchion supports attached.
  4. Retract the lower superconducting coil to a position below the TF coil leg.
  5. Remove four sets of intercoil support structure between three adjacent TF coil legs.
  6. Demount and remove three adjacent outboard TF coil legs.
  7. Remove the shield modules in front of the sector.
  8. Remove the two poloidal divertor modules.
  9. Uncouple the vacuum duct from the bottom surface of the sector.
  10. Cut the internal vessel joint.
  11. Engage a sector handling device and extract the sector.
-

As mentioned previously, reassembly of the sector will be more difficult and time-consuming because of the requirements for alignment, preparation of weld surfaces, and intermediate tests for vacuum, coolant, electrical, and structural integrity.

REFERENCES

1. *Fusion Engineering Device*, DOE/TIC-11600, October 1981.
2. C. A. Flanagan, D. Steiner, and G. E. Smith, "*Fusion Engineering Device Design Description*," ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee, December 1981.

10. COST AND CONSTRUCTION SCHEDULE

10.1 FED-R CAPITAL COST PROJECTION

The initial cost for FED-R is estimated for the Stage I machine. The tokamak and associated systems are designed for Stage II performance and are de-rated for Stage I operation. Only the required amount of power supplies, heat transport equipment, and plasma heating equipment necessary for Stage I operation are included in the Stage I cost. It is assumed that this equipment can be installed in modules, with additional modules installed later for upgrading the device to Stage II.

The FED-R parameters for Stages I and II as used in the FEDC systems code are presented in Table 10-1. The capital cost of Stage I, based on those parameters, is shown in Table 10-2. The capital cost estimate is based on unit cost values from fusion projects such as TFTR, TSTA, LCP, and JET and from data supplied by architectural engineering firms. The direct capital cost for Stage I FED-R is estimated to be 810 M\$. The indirect cost, including 45% for engineering and 15% for installation, is 500 M\$ and the total (direct and indirect) cost is 1300 M\$. Applying a 30% contingency brings the total direct cost for Stage I FED-R to 1690 M\$.

Stage II FED-R cost consists of the Stage I value plus the cost of additional power supplies, energy storage equipment, heat transport equipment, and additional plasma heating equipment. Table 10-3 shows that the additional cost for these items for the Stage II upgrade is 220 M\$ in direct capital cost. Applying the above mentioned factors for the indirect costs and contingency, the total capital cost increment for the Stage II upgrade is 450 M\$.

10.2 CONSTRUCTION SCHEDULE PROJECTION

The preliminary construction schedule projection for FED-R is based upon the schedule which was prepared for the FED baseline design.^{1,2} The FED baseline schedule was developed by first estimating design, procurement, fabrication, and installation times for each major component and system and then establishing an installation sequence which is

Table 10-1. System code simulation

FED-R parameters				
Parameter	Unit	Value		
		Stage I	Stage II	
Geometry				
Major radius, R	m	3.5		
Minor radius, a	m	0.85		
Scrape-off layer	m	0.10		
Plasma elongation, κ	-	1.5		
Aspect ratio, A	-	4.12		
Plasma				
Safety factor, q_ψ		2.25	2.25	
Plasma current	MA	3.5	4.4	
Effective charge, Z_{eff}	-	1.5	1.5	
Ripple (peak-average), edge	%	1.0	1.0	
Total beta, β	%	5.0	5.5	
Field on axis, B_T	T	4.0	5.0	
D-T fusion power, P_{th}	MW	75	250	
Fusion power gain, Q	-	1.5	2.5	
Neutron wall loading (outboard wall near midplane)	MW/m ²	0.40	1.34	
Bulk shield				
Inboard shield thickness	m	0.45		
Outboard shield thickness	m	1.1	Between TF coils	
TF coil				
Number of TF coils	-	12		
Maximum TF field	T	7.0	8.75	
Current density, J	A/cm ²	650 inside leg 330 outside leg	815 415	
I ² R loss (TF)	MW	180	280	
TF power (startup)	MW	200	308	
Total amp turns	MAT	70	87	
Coil current	kA	400	500	
Charge time	s	600	600	

Table 10-1 (continued)

FED-R parameters			
Parameter	Unit	Value	
		Stage I	Stage II
<i>PF coils</i>			
EF coil amp turns	MAT	15	19.5
OH solenoid amp turns	MAT	45	45
Total PF amp turns	MAT	60	64.5
V-s from EF coils	Wb	9.2	11.9
V-s from OH coils	Wb	10.5	10.5
Total V-s from PF	Wb	19.7	22.4
Current rise time	s	20	20
Current density EF coils (winding pack)	A/cm ²	1200 inner coils/1120 outer coils	1500 1400
Field, OH solenoid	T	9.4	9.4
Current density OH coils (winding pack)	A/cm ²	3750	3750
I ² R loss [FF]	MW	39	63
I ² R loss [OH] *	MW	142	142
Copper coil current	kA	96	120
Superconducting coil current	kA	40	50
<i>Plasma heating</i>			
Number neutral beams	-	3	4
Injected power	MW	50	100
Beam energy	keV	150	250
Overall beam efficiency	%	42	70
ECRH injected power	MW	2	2
ECRH frequency	GHz	93	93

* Peak value at beginning of each pulse.

Table 10-2. FED-R cost estimate summary

Stage I		
1. Magnet System		191.9
TF coil	147	
PF coil	35	
Intercoil structure	8.9	
Bucking cylinder	1.0	
2. Torus		120.6
First wall	8.4	
Armor	1.2	
Shield	87	
Divertor module	20.2	
Torus Support	3.8	
3. Cooling Systems		25
Heat transport loops	14	
Cooling Tower	11	
4. Tritium and Fuel Handling		48.2
Primary fuel cycle	6.4	
Secondary systems	16.0	
Tritium system data acquisition	9.1	
Tritium cleanup (Room)	12.7	
Fuel injector	4.0	
5. Plasma Heating		74.2
Bulk heating (NBI)		
NBI	58	
Shielding	6.6	
Pre-heating (ECRH)	9.6	
6. Electrical Systems		89.6
PF Electrical	57	
TF Electrical	11	
AC Power	21.6	
7. Vacuum Pumping System		10.6
Vacuum ducts	6.4	
Vacuum pumps	4.2	

Table 10-2 (continued)

		Stage I	
8.	I&C		47
	Diagnostics	22	
	Information & control systems	25	
9.	Maintenance Equipment		60.4
	Reactor Cell	33.3	
	Hot cell	27.1	
10.	Facilities		142
	Reactor cell	35.5	
	Hot cell	35	
	Cooling system structures	1.1	
	Cryogenic refrigeration bldg.	1.0	
	Radiation waste bldg.	4.3	
	Administration bldg.	4.3	
	Mockup & shop bldg.	13.5	
	Power supply & energy storage bldg.	12.8	
	Diesel generator bldg.	0.4	
	Tritium processing bldg.	8.1	
	Ventilation bldg & stack	13.7	
	Site improvements	12.3	
	TOTAL DIRECT COST		810
11.	Indirect Costs		
	Engineering & Mgmt. (45%)		365
	Installation (15%)		122
	Total (Direct + indirect)		1297
	Contingency (30%)		389
	TOTAL COST		1686

Table 10-3. Cost of added capacity for Stage II operation

Component	Stage I cost	Stage II cost	Cost increment
Neutral beams*	65 M\$	189 M\$	189 M\$
PF electrical	57	66	9
TF electrical	11	15	4
AC power	22	26	4
Heat transport	25	36	11
Total cost increment			217

* Beams are not upgraded but completely replaced in Stage II operation.

compatible with the foregoing estimated span times. As pointed out in Ref. 1, little directly applicable experience exists in carrying out an endeavor having the unique features and level of complexity exhibited by the FED project. The FED construction schedule and installation sequence estimates have, therefore, relied heavily on the judgment and experience of the Design Center personnel and also on estimates made for other major construction activities, both within and outside the fusion energy community.

The FED baseline construction schedule is predicated on the following ground rules regarding conditions at the start of the schedule:

- The beginning of preliminary design serves as the starting point for the schedule (i.e., the start of Title I).
- A satisfactory conceptual design has been completed and approved.
- Major contractors (e.g., project management, construction management major component design organizations, and the architect and engineering firms) have been selected and are ready to begin work.
- The site has been selected and the environmental impact statement has been completed and approved.

These ground rules are generic in nature and would be applicable to FED-R.

The overall construction schedule, as developed for the baseline, is divided into five phases. In Phase I, a preliminary design is carried out and appropriate preliminary design reviews are conducted, the site is prepared, the safety review is completed, and construction approval is received. Component design and procurement (in particular for long lead items) are performed in parallel with the aforementioned preparatory steps. The preparatory phase is estimated to be 24 months for the FED baseline.¹ The Phase I span time of 24 months is also considered to be a reasonable projection for FED-R, since the design complexity, site size, and safety considerations are quite similar to the FED baseline device.

The second phase starts with actual construction activities on the tokamak building. During this phase, the tokamak building construction

is taken to the point where device component installation can begin. The baseline tokamak building construction phase is expected to take approximately 21 months. As noted in Section 8 (Facilities Description), the FED-R reactor building volume is determined to be approximately 10 percent less than the baseline reactor, due primarily to a reduction in building height. However, the floor area of the building increases to accommodate larger heat exchangers and other cooling equipment needed for cooling of the copper TF coils. The impact on building size due to this change has not been evaluated, but this is likely to make the FED-R reactor building the same size as the FED baseline building. Accordingly, it is assumed that the FED-R reactor building will also require a 21 month period for construction.

Direct comparison at an overview level of FED-R and the FED baseline design with respect to Phases 3 and 4 of the construction schedule was found to be impractical since the two devices are different in design approach and, therefore, the order of assembly of major components and the associated schedule logic are also different. As a consequence, preparation of a preliminary construction schedule projection (specifically for FED-R) is undertaken.

The first step is to determine the time required to carry out designs, negotiate procurements, and complete fabrication of each of the FED-R major components. This is done by comparing the major FED-R components to similar FED baseline components and determining, with the aid of the cognizant FEDC engineers, whether the manufacturing time span for a given component should be longer, shorter, or the same as that of its FED baseline counterpart.

The next step is to establish the required order of assembly of the FED-R major components and establish from this information a construction schedule logic. Having established the assembly logic, it is next necessary to determine the time required for installation of each major component. As in the case of the component manufacturing time, comparison with the baseline installation time spans, and consideration of the relative complexity of the components provided the basis for determining whether the installation time for a given FED-R component should be longer, shorter, or the same as its baseline counterpart. In addition

to the definition of time spans, it is also determined what installation tasks could be accomplished in parallel and also where an overlap between related tasks could be accommodated. The motivation here is to reduce the schedule time where possible.

Once the schedule is laid out using the appropriate logic, time spans, and overlaps, it is necessary to check that each component would be available for installation when required. This is done by referring back to the manufacturing time spans of each component and determining whether component manufacturing completion dates met the required need dates.

The resulting schedule projection is shown in Fig. 10-1. The milestones shown at the top of the schedule were used for the FED baseline¹ and are generic in nature. The Phase I preliminary design effort and associated tasks (preliminary safety analysis report, site investigation, etc.) are expected to require the same time span as the baseline (24 months) as indicated earlier. Building fabrication starts at the beginning of the third year. The schedule specifically addresses the fabrication of the reactor building and the assembly of the device components within the building. These elements of the overall schedule are of particular interest in that they are on the project critical path. The reactor building fabrication time of 21 months (prior to installation of the first component) is expected to be the same as the baseline, as mentioned earlier.

Installation of components starts 45 months into the program with emplacement of the lower section of the bucking cylinder. Additional major components are installed in the following order:

- TF coil inner legs
- Lower internal copper EF coils (3)
- Platform (at the bottom of the torus)
- Inboard, top, and bottom shields
- Upper internal copper EF coil (1)
- First wall and vacuum vessel
- Neutral beam drift tubes (3)
- Outboard shield (partial installation)

Fig. 10-1. FED-R preliminary construction schedule projection.

- TF coil outer legs
- Intercoil structure
- Remainder of outboard shielding (between TF coils)
- Lower and upper superconducting PF coils

As in the case of the FED baseline, four additional months (after installation of the superconducting PF coils) are allowed for completion of installation of piping, mechanical equipment, instrumentation, electrical equipment, cables, and cable trays. These activities are initiated at the time of first component installation and are continued in parallel throughout the installation period. Also required, but not shown in the schedule, is the installation of peripheral equipment (pumps, valves, fuel injectors, etc.) — based on experience with the FED baseline schedule, it is not anticipated that these items will be schedule drivers.

Integrated system check-out is the final construction phase prior to the start of Stage I testing. Six months time is allocated for this activity.

The total time span from go-ahead to completion of preoperational testing for FED-R is projected to be 92 months. It must be noted that the schedule as shown allows no time for contingencies or changes and should, therefore, be considered as a lower bound on schedule time span.

Schedule elements anticipated for the FED-R operational period, including the integrated system check-out, are summarized as follows:

• Pre-nuclear testing	
-Phase A, integrated systems checkout	0.5 y
-Phase B, tritium-free operation	1.5 y
-Phase C, tritium operations checkout	1 y
• Stage I, nuclear testing, low wall loading	6 y
• FED-R upgrade for Stage II	1.5 y
• Stage II, nuclear testing, high wall loading	9.5 y
Total FED-R operation schedule time span	20 y

REFERENCES

1. *Fusion Engineering Device*, DOE/TIC-11600, October 1981.
2. C. A. Flanagan, D. Steiner, and G. E. Smith, *Fusion Engineering Device Design Description*, ORNL/TM-7948/Vol. I and II, Oak Ridge, Tennessee, December 1981.

11. R&D REQUIREMENTS

Successful design and construction of FED-R would require implementation of R&D on several key engineering, safety, and environmental issues. Discussion in this section is aimed at engineering R&D needs. The main objective of these R&D tasks is to develop a basis for those technological features that appear to be beyond the current state of the art.

11.1 CERAMIC INSULATOR FOR THE COPPER TF COILS

The large planar faces between turns in the normal copper TF coils of the FED-R are insulated by a ceramic coating. The ceramic specified is the spinel MgAl_2O_4 , applied to both contacting faces by plasma spraying. Thin layers are used to provide maximum adherence and adequate insulation and to occupy minimum space.

There is a significant industrial experience with MgAl_2O_4 , and representative properties are available in handbooks. The data base on MgAl_2O_4 is summarized in Appendix D. Experimental studies have shown that under irradiation the material is resistant to swelling and is strengthened by high fluence irradiation at temperatures somewhat above those of the proposed application. While spinel has been deposited on metal substrates, it has probably not been applied with the large area coverage proposed for the FED-R magnets. In addition, the properties of MgAl_2O_4 and of most other ceramics are dependent on the specific product and on production variables. This requires a development and testing program to qualify the material for each particular application.

Qualification of plasma-sprayed MgAl_2O_4 for use on the TF coils requires a development program to examine the effect of material and production variables on critical properties. Material variables include stoichiometry of the MgAl_2O_4 ; controlled additives, such as TiO_2 , used to modify the deposit properties; thickness of deposit; and porosity of the coating. Processing variables must include surface preparation of the copper substrate, rate of deposition, and other deposition variables. Techniques for assuring the adequacy of coating must be developed, as

must techniques to repair defective areas. (A dc voltage probe can probably be used for inspection, and defective areas may be removed by grinding and replaced by respraying.)

Materials testing is required to assure the adequacy of the spinel insulation in meeting the magnet performance requirements. The dominant requirement is the maintenance of insulating capability, which could be influenced by cracking, crazing, or spalling of the coating. Appropriate simulation of the service conditions can be used, including irradiation of copper-cladding samples in fission reactors and flexure testing to simulate magnet cyclic loading. Testing during and after loading will include macroscopic inspection for delamination, spalling, or cracking and conductivity measurement with dc voltage probes.

SPINEL may also be suitable for use as the radiation-resistant insulator in the internal poloidal field coils. A modified method of applying the insulator may have to be developed. The insulator will be subjected to different types of strains than in the TF coils, so that pretesting will be necessary.

11.2 TF COIL FABRICATION

The FED-R TF coils are much larger than any previously built copper coils, and simple extrapolation of existing copper-coil technology (e.g., as used in PDX, DIII, or ISX) to the FED-R coils may not be possible. In particular, development work is required to ensure the fabricability of copper plates of the required thickness, width, and length that will also maintain satisfactory mechanical and electrical properties.

Fabrication of the TF coil segments making up the inboard legs requires bending 6-cm thick copper plate material to accommodate the cross-section transition from rectangular to trapezoidal. The fabrication also requires machining these plates to a tapered thickness while maintaining tight dimensional tolerances in order to obtain acceptable fitup in the nose region. The ability to hold such tolerances during manufacture must be demonstrated.

The feasibility of the demountable finger joints must be verified, and a significant step in this direction will be taken with the disassembly of the DIII TF coils in 1984 for installation of a new vacuum vessel. Development is also desirable to identify and prove out an alternate demountable joint concept that would permit faster disassembly and reassembly than the reference concept adopted herein.

11.3 NEUTRAL BEAM INJECTORS

The principal advancement required for the Stage I positive-ion-based neutral beam injectors beyond the performance of the TFTR injectors is the capability for steady-state operation. At present, 80-keV beams with 30-s pulse lengths and relatively low duty factor are being developed for the MFTF-B, but an additional program must be initiated to insure actual steady-state neutral beam operation.

The present development status of the components of the Stage II negative-ion-based neutral beam injectors is as follows: The negative-ion source program at LBL has produced more than 1 A of D^- current in multisecond pulses, compared with the 20 A per source required in the FED-R injectors. The transverse-field-focusing accelerator will be tested for ion beam acceleration at LBL in the mid-1980s. An oxygen-iodine chemical laser of the type to be used for photodetachment neutralization has been demonstrated at the Air Force Weapons Lab. LBL plans to operate a low-current prototype neutral beam of the FED-R type by 1989.

The present LBL D^- source has a current density of only 10 mA/cm². Additional development is required to obtain the current density of at least 30 mA/cm² that is needed to produce the 100-A beam in each beam duct of FED-R.

11.4 VACUUM VESSEL

A significant program is required to develop and prove out practical remote maintenance techniques and specialized equipment for purposes such as the disassembly and replacement of vacuum vessel sectors and of

divertor modules. Techniques for rapid remote cutting and welding on the vessel are required to insure the feasibility of the sector joints.

The vessel wall and divertor collector plates must dispose of extremely high thermal fluxes in steady-state operation. Materials that can withstand these thermal loads as well as the mechanical stresses experienced in tokamak operation are being developed on test stands in the DOE/ANL first-wall program and will be proof-tested in the large tokamaks that will operate in the mid-1980s (TFTR, D-III Upgrade, JET, and JT-60).

APPENDIX A

FED-R PARAMETER LIST AND SPECIFICATIONS

A-1 Introduction

This document provides specifications for the Fusion Engineering Device with Resistive Toroidal Field Magnets (FED-R). Its primary purpose is to establish design guidelines based on systems performance requirements.

The primary mission of the FED-R is to provide a reliable, quasi-steady 14-MeV neutron source of the order of 100 MW intensity and irradiating at least several tens of square meters of test module area for the purpose of developing fusion-neutron applications such as tritium and fissile breeding and electrical power production, as well as radiation effects testing.

A-2 Performance Objectives

The performance goals of FED-R have been established as follows:

- 1) fusion neutron wall loading $\geq 0.4 \text{ MW/m}^2$ over at least 50 m^2 test area;
- 2) fusion energy gain $Q_p \geq 1.0$;
- 3) accessibility for easy change-out of nuclear test modules, loops, and other assemblies of damaged reactor components and for maintenance;
- 4) high availability ($\sim 10^7$ burn-s/yr);
- 5) steady-state operation, when required, to be achieved with non-inductive current drive; and
- 6) radiation level of 5 mr/h or less outside of the reactor 36 h after shutdown.

A Q_p of the order of one or larger is required to minimize the capital cost and power requirements of the plasma heating equipment, as well as to ameliorate the heat removal problems in the torus.

A-3 Principal Features

The FED-R is a tokamak fusion reactor with the following principal features:

Toroidal Field Coils	Normal conducting copper
Poloidal Field Coils	Normal copper coils inside TF coil bore, superconducting coils outside TF coil bore
Helium Ash and Impurity Control	Expanded-boundary single null poloidal divertor
Bulk Heating	Neutral beams (RF back-up)
Startup Technique	ECRH assist, OH and current drive
Plasma Chamber	Six 60° sectors — radially removable
Plasma Chamber Vacuum Boundary	Inside bulk shield
Fueling System	Combination D and T pellet injectors and neutral beams
Plasma Chamber Vacuum Pumping	Poloidal divertor

A-4 Design Control Parameters and Baseline Parameters

The design control parameters are listed in Table A-1, and they provide the basis for the baseline parameters listed in Table A-2.

A-5 Schedule

Table A-3 shows the plan of operations over the first three years operating life. Table A-4 describes two stages of D-T operation for 17 years of nuclear testing. The 17 years includes a 1.5 year shutdown period between Stages I and II needed for upgrading certain subsystems.

A-6 Critical Engineering Areas

Table A-5 lists the major critical engineering areas for FED-R. Preliminary concepts for the critical design problems have been developed in the course of this study and are described in the main body of this report.

Table A-1. Design Control Parameters

Description	Unit	FED-R Value	
		Stage I	Stage II
IMPLEMENTATION DATE		1990	1998
<i>Geometry</i>			
Major radius, *R	m	3.5	3.5
Plasma radius, *a	m	0.85	0.85
Plasma elongation, *κ	m	1.5	1.5
Plasma triangularity, *δ		0.25	0.25
Radial build	(See Figure 2)		
Field on axis	T	4.0	5.0
Plasma current	MA	3.5	4.4
<i>Plasma</i>			
Average ion temperature, $\langle T_i \rangle$	keV	15 ^Δ	20 ^Δ
Safety factor (edge), q_ψ (flux surface averaged)		2.25	2.25
Effective charge (during burn), Z_{eff}		1.5	1.5
TF ripple (peak-to-average), Edge	%	1.1	1.1
Total beta, $\langle \beta \rangle$	%	4-6	4-6
Fusion gain, Q_p		1.5	2.5
<i>Operating Mode</i>			
Burn time, t_{burn}	s	10,000	28,000
Cycle time, T_{cycle}	s	10,250	28,250
Pumpdown time, t_p	s	200	200
Startup/shutdown time, t_{ss}	s	20/20	20/20
Number of pulses/year		1,000	350
Lifetime	Years	6	9.5
Lifetime burn time	Ms	60	93
Lifetime 14-MeV fluence on outboard wall	MW-Yr/m ²	0.77	3.6

* See Fig. A-1

^Δ Average electron temperature (T_e) = 0.5 T_i

Table A-1. Design Control Parameters (continued)

Description	Unit	FED-R Value	
		Stage I	Stage II
Toroidal Field Coils			
Number		12	12
Peak design field at winding, B _m	T	7.0	8.75
Clear bore height	m	7.2	7.2
Clear bore width	m	3.7	3.7
Stored energy per coil	MJ	300	470
I ² R loss (Goal)	MW	175	270
Charge time	s	600	600
Poloidal Field Coils			
Volt-seconds (total)		30	30
EF		10	10
OH		20	20
Blip coils		TBD	TBD
OH charge up time	s	20	20
OH discharge time (during start up)	s	20	20
OH current ramp down time	s	100	100
OH pulse duration	s	140	140
Time average power (resistive)	MW	1	1
Plasma Heating			
Start up			
Initiating voltage	V	25	25
Current rise time	s	20	20
Vacuum vessel resistance	μΩ	100	100
ECRH			
Time duration	s	2	2
Power	MW	2	2
Bulk heating			
Neutral beam	keV	150	250
Ion. base		D ⁺ Δ	D ⁻ Δ
Injection angle	degrees	45	45

Table A-1. Design Control Parameters (continued)

Description	Unit	FED-R Value	
		Stage I	Stage II
Plasma Heating (continued)			
Pulse length	s	10,000	28,000
Power			
Initial	MW	50	100 ^Δ
Burn	MW	50	100 ^Δ
Input power	MW	140	140
Ion species (D ⁺ , other)	%	90, 10	100
Number of injectors		3	4
Efficiency		~0.36	~0.72
Number of pellet injectors		2	2
Pellet velocity	m/s	2000	2000
Injection rate			
Startup	s ⁻¹	TBD	TBD
Burn	s ⁻¹	TBD	TBD
Pellet diameter	mm	4	4
Pellet composition			
Deuterium	%	>90	>90
Tritium	%	>90	>90
Mixed (T/D ratio)		0.5	0.5

^Δ Injectors in Stage II replace injectors in Stage I.

Table A-1. Design Control Parameters (continued)

Description	Unit	FED-R Value	
		Stage I	Stage II
First Wall/Armor			
Coolant — Inboard		Water	Water
— Outboard		Helium	Helium
Outboard neutron wall load (ϕ_w)	MW/m ²	0.4	1.3
Average radiation heat load	W/cm ²	10	25
Total charge exchange power load to first wall	MW	TBD	TBD
Charge exchange particle energy	eV	TBD	TBD
Total charge exchange particles	s ⁻¹	3.1 × 10 ²²	3.1 × 10 ²²
Deposition area for charge exchange particles (includes limiter surface)	m ²	TBD	TBD
Shield			
Inboard thickness (excluding spool, armor, gaps, scrapeoff)	cm	39	39
Outboard thickness behind TF coils	cm	19	19
Time after shutdown to permit personnel access	h	36	36
Level of radiation			
—at outer boundary of reactor	mrem/h	5	5
—at TF coil joint	mrem/h	20	20
Outboard thickness between TF coils	cm	90	90
Maximum structure temperature	°C	300	300
Vacuum			
Initial base pressure	torr	10 ⁻⁷	10 ⁻⁷
Preshot base pressure	torr	10 ⁻⁵	10 ⁻⁵
Postshop base pressure	torr	3 × 10 ⁻⁴	3 × 10 ⁻⁴
Pressure at duct inlet during burn	torr	10 ⁻²	10 ⁻²
Particle flux (molecular) to be pumped	s ⁻¹	TBD	TBD

Table A-2. Baseline parameters of FED-R

Description	Unit	Value				
<u>Toroidal field coils</u>						
Number		12				
Peak field on winding	Tesla	8.75				
Ampere-turns per coil	MAT	7.3				
Winding current	kA	360				
Winding current density	A/cm ²	1000				
Overall current density	A/cm ²	800				
Stored energy/coil	MJ	470				
Conductor type		Normal				
Conductor cooling		Water				
Winding insulation		Spinel				
Coil vertical bore	m	7.2				
Coil horizontal bore	m	3.7				
Radial build	m	1.3 inboard 1.7 elsewhere				
Out-of-plane build	m	1.04				
Conductor weight/coil	kg × 10 ³	368				
Structural weight/coil	kg × 10 ³	49				
Total weight/coil	kg × 10 ³	417				
<u>Poloidal field coils</u>						
	<u>Unit</u>	<u>On solenoid</u>	<u>Top normal coil</u>	<u>Bottom normal coil</u>	<u>Top SC coil</u>	<u>Bottom SC coil</u>
Peak field on winding	T	12			1.2	4.1
Ampere-turn/coil	MAT		2.2	2.5 (each)	2.0	9.8
Winding current	kA		120	120	50	50
Winding current density	A/cm ²		1400	1400	1400	1200
Overall current density	A/cm ²		1200	1200	500	450
Stored energy/coil	MJ		21	103	83	1700
Conductor type		Normal	Normal	Normal	SC	SC
Conductor cooling		Water	Water	Water	Helium	Helium
Winding insulation		TBD	TBD	TBD	G10	G10
Current ramp time	s		20	20	20	20
Coil inside radius	m	0.5	2.3	2.8	7.8	7.5
Radial build	m	0.2	0.4	0.4	0.4	1.0
Axial build	m	6.0	0.4	0.4	0.4	1.0
Conductor weight/coil	kg × 10 ³		20	74	23	110
Structural weight/coil	kg × 10 ³		2	9 (all)	21	99
Total weight/coil	kg × 10 ³		24	83	44	209

See Fig. A-3 for PF coil locations

Table A-2. (continued)

Description		Stage I	Stage II
<u>Neutral beams</u>			
Number of injectors		3	4
Drift tube length	m	~6	~6
Shield thickness	m		
Shield weight	Tons		
Magnetic shield thickness	m	0.15	0.15
Magnetic shield weight	Tons	90	130
Injector weight	Tons	60	75
Injector size (W,H,L)	m	4 x 6 x 5	7.5 x 7.5 x 5.2
<u>ECRH assist</u>		<u>Unit</u>	<u>Value</u>
Number of launchers			12
Launching position			High field side
Launching mode			Ordinary
Shield thickness	m		( )
Magnetic field at startup position	T		3.3
Number of operating frequencies			1
<u>Fueling - pellet injectors</u>			
Number			2
Location	Km/s		2
Pellet size	mm		4
Pellet rate (D/T)	s ⁻¹		0-15 (Stage II)
Distance from plasma	m		5
<u>Fueling - gas puffers</u>			
Number of systems			2
Number of nozzles			12
Location			Between sectors
Maximum gas flow rate (D/T)	g/s		0.5

Table A-2. (continued)

Description	Unit	Value
<u>First wall/armor</u>		
<u>Shield</u>		
Inboard shield thickness	m	0.39
Inboard shield weight	Tons	
Outboard shield thickness	m	0.19
Outboard shield weight	Tons	
Maximum shield temperature	°C	200
Shield material		Nitronic 33
<u>Vacuum</u>		
Pumping speed	Torr l/s	
Active pumping area	m ²	
Total pumping area	m ²	
<u>Tritium</u>		<u>Stage II</u>
Maximum consumption	g/h	1.6
Inventory		
Injectors	g	2.5
Fuel processing	g	350
Storage	g	1150
Complete fuel cycle	g	1525
<u>Reactor building</u>		
Building volume	m ³	131,000
<u>Plant cooling</u>		
Total plant load	kW	
Nature of heat rejection		Wet cooling tower
<u>Radiated waste</u>		
Method of off-site shipment		Sealed container, truck
Number of shipments	yr ⁻¹	

Table A-3. FED-R Baseline plan of operations (pre-nuclear testing)

Phase	Period (after start up) years	Description	Peak toroidal field	Operating mode (on average)	Number of pulses
A	0-0.5	Integrated systems checkout	≤ 6 T	6 days/week 2 shifts/day 1 pulse/10 min Downtime of 2 weeks/month	7,500
B	0.5-2.0	Tritium-free operation, 100 s to 10,000 s pulses	≤ 6 T	6 days/week 2 shifts/day Downtime of 1 week/month	7,500
C	2.0-3.0	As per Phase B, including tritium fueling and recycling systems			

The above phases are followed by Stages I and II of Nuclear Testing described in Table A-4.

Table A-4. FED-R D-T operations

There are 2 stages of D-T operation planned for FED-R.

	<u>Stage I</u>	<u>Stage II</u>
Period after start-up	3 to 9 yrs. Low wall loading Nuclear testing	10.5 to 20 yrs. High wall loading Nuclear testing
Fusion power gain, Q_p	1.5	2.5
14-MeV neutron wall loading at outboard test modules	0.4 MW/m ²	1.3 MW/m ²
Peak field at TF coils	7.0 T	8.75 T
Pulse length	10,000 s	28,000 s
Number of pulses	1,000/yr	350/yr
Duration	6 yr	9.5 yr
Total pulses	6,000	3,325
Total 14-MeV neutron fluence	0.8 MW-yr/m ²	3.9 MW-yr/m ²

There is a downtime of 1.5 yrs. between Stage I and Stage II for upgrading certain subsystems.

Table A-5. FED-R critical engineering areas

Priority	Status
1. Design demountable TF coil joints	Preliminary concept developed
2. Design vacuum vessel with resistance $\sim 0.1 \text{ m}\Omega$, adequate heat removal, and ease of disassembly	Completed
3. Design internal PF coils with demountable joints and determine feasible replacement procedure	Completed
4. Minimize first-wall material between the plasma and test modules	Completed
5. Determine procedures for 24-h replacement of significant fraction of test modules (after 36-h cooldown)	Preliminary
6. Determine mechanical properties of SPINEL insulator in suitable form for plate-type TF coils	Open
7. Develop 250 keV neutral beam injectors based on negative ions (Stage II)	Addressed by LBL Program

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$$R = 3.5 \text{ m}$$

$$a = 0.85 \text{ m}$$

$$K = \text{ELONGATION} = b/a = 1.5$$

$$\delta = \text{TRIANGULARITY} = c/a = 0.25$$

Fig. A-1. Plasma dimensional characteristics.

*ALL DIMENSIONS IN cm

Fig. A-2. FED-R radial buildup.

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Fig. A-3. TF and PF coil configuration.

APPENDIX B

TOKAMAK DESIGN INTERACTIVE CODE^{*}

Scoping studies of FED-R were aided by the development and use of the TORFA computer code for drawing elevation, plan, and radial views of rectangular coil tokamaks. The code allows the effects of dimensional changes of parameter to be seen immediately in the drawings and also in the values of dependent parameters, such as the area available for blankets and the required length of the neutral beam drift ducts.

The tokamak drawings are constructed from input parameters and dimensions that are subsequently available for interactive manipulation. Table B-1 is a sample listing of input parameters, which include the number of TF and PF coils, their dimensions and locations, dimensions of the plasma, vacuum vessel, blanket and shield sectors, beam injector ducts, and injector angles.

Output flexibilities include display of components individually (i.e., TF system components, PF systems, vacuum vessel, shield components, blanket sectors) or in combination; rescaling (to provide blow-ups); and display of 1, 2 or 4 quadrants in plan view. Table B-2 is a dialogue listing indicating output flexibility.

Figures B-1 through B-4 show typical output drawings of the TORFA code.

---

^{*}This code was developed by L. D. Stewart.

Table B-1. Sample listing of input parameters

TIT = FED-R (title)

TF system parameters:

NCOLS = 12 (number of toroidal field coils)

THKIT = 1.28 m (toroidal field coil inner leg thickness)

THKITF = 1.70 m (toroidal field coil upper, lower & outer leg thicknesses)

UTF = 1.04 m (toroidal field coil width)

JOINTS = 2 (number of joints per TF coil)

THKLJ = 0.44 m (TF coil lap joint thickness)

DLJ = 2.50 m (distance from TF coil outside corner to end of lap joint)

RADCP = 0.72 m (center post radius)

PF and OH parameters:

Radial locations of upper/lower poloidal field coils:
 (< 0 positions inside TF coil; > 0 outside; = 0 deletes)

PFY(1) = -2.60 m PFY(2) = -3.60 m

HPYU = 0.40 m (height of upper/lower poloidal field coils)

HPYCO = 2 (number of outer poloidal field coils)

Vertical displacements of outer poloidal field coils from tokamak midplane:

PFY(1) = 3.30 m, positioned outside TF

PFY(2) = -3.50 m, positioned outside TF

THKPO = 1.00 m (thickness of outer poloidal field coils)

THKONS = 0.30 m (thickness of OH solenoid)

< 0 positions inside TF coil; > 0 outside)

HONS = 6.00 m (height of OH solenoid)

Table B-1 (continued)

Plasma, vacuum vessel, blanket module and shield parameters:

ELONG = 1.60 (plasma elongation)
 THKVAC = 2.05 m (vacuum vessel radial extent)
 HPMPCB = 1.10 m (pumping chamber height)
 THKIB = 0.00 m (inboard blanket thickness)
 THKOB = 0.80 m (outboard blanket thickness)
 THKUB = 0.00 m (upper (and lower) blanket thickness)
 THKIS = 0.48 m (inboard shield thickness)
 THKOS = 0.30 m (outboard shield thickness)
 THKUS = 0.80 m (upper (and lower) shield thickness)
 THKOG = 0.08 m (gap thickness: outboard shield to TF coil)
 THKUG = 0.40 m (gap thickness: upper (or lower) shield to TF coil)

Heating system parameters:

AINJ1 = 42 deg (injection angle at plasma outer radius of injector set 1)
 NINJ1 = 3 (number of injectors in set 1)
 AINJ2 = 20 deg (injection angle at plasma outer radius of injector set 2)
 NINJ2 = 0 (number of injectors in set 2)
 DUCTU = 0.80 m (injector duct width)
 DUCTH = 1.05 m (injector duct height)
 SDUCTF = 0.20 m (separation between duct wall and TF coil)

x

Table B-2. Dialogue listing indicating output flexibility

Type E to draw Elevation view
 Type R to draw Radial view
 Type P to draw Plan view
 Type F to Format drawing
 Type L to List input parameters
 Type parameter name to change input parameters
 Type I to re-Initialize input data
 Type U to begin Versatec drawing file output
 Type LO to List Output (computed) data
 Type U to Write output data file

X/

Label or scale modifications? yes

Date?

Title?

Labels?

Change number of plan view quadrants displayed?

Scale change?

Draw individual components? yes

Center post & TF clamps?

Toroidal field coils? yes

All TF components?

Inner TF coil leg?

Outer TF coil leg?

Shadow of TF coil top (plan view)?

Vacuum vessel?

Shielding? yes

All shielding components?

Upper shielding block?

Lower shielding block?

Inner shielding block?

Outer shielding block?

Outer shielding plug?

Blanket region? yes

Total blanket region?

Upper blanket region?

Lower blanket region?

Inner blanket region?

Outer blanket region?

Poloidal coils? yes

Total poloidal coil system?

Upper poloidal coils?

Lower poloidal coils?

Outer poloidal coils?

OH solenoid?

Plasma?

Injectors?

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Fig. B-1. Typical TORFA code drawing.

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Fig. B-2. Typical TORFA code drawing.

Fig. B-5. Typical TORFA code drawing.

Fig. B-4. Typical TORFA code drawing.

APPENDIX C

FED-R STAGE II ALTERNATIVE

The FED-R implementation plan envisages a 2-stage operating program. Stage II specifications call for higher magnetic field, plasma current, and neutral beam power, but with the same tokamak configuration as in Stage I. However, the ability to replace the outboard legs of the TF coils of FED-R permits a relatively inexpensive conversion of the test reactor into an industrially useful demonstration plant of at least 250 MW fusion power, for purposes such as breeding fissile fuel or other practical applications of fusion neutrons. The converted reactor would have nearly the same field, current, and neutral-beam power as planned for the reference Stage II version discussed in the body of this report.

In the following it is assumed that the horizontal bore of the TF coils is to be increased by 1.0 m in order to accommodate a complete-coverage blanket and a slightly larger plasma than in FED-R Stage I. This conversion would be accomplished by the following steps:

- 1) Disassemble the TF coil joints and retract all the outboard coil legs.
- 2) Replace the outboard legs with new legs whose horizontal arm segments are 1.0 m longer. *Alternatively*, insert segments of 1-m length between the top and bottom arms and the original outboard legs. The innermost end of each segment would be joined permanently to the stationary section of each coil.
- 3) The horizontal bore of the TF coils will now be 1.0 m larger than in FED-R. The increase in the TF coil power dissipation will be 5% for a given magnetic field at the inboard coil legs.
- 4) Replace the FED-R vacuum vessel by one of approximately 0.7-m larger major radius and 0.1-m larger minor radius.
- 5) Install a complete "wraparound" blanket assembly. The 40-cm inboard shielding would be replaced by a blanket approximately 1 m thick.
- 6) The internal PF coils would be retained but their currents would be reprogrammed. An additional pair of internal PF coils at larger major radius would be installed.

- 7) The external PF coils could be retained if they are initially made oversized (by 2 m diameter) for FED-R Stage I.

If conversion of FED-R into a demonstration reactor in Stage II were anticipated during the FED-R design phase, then the siting of the FED-R facility would be selected in part according to the envisaged ultimate application. This plan would insure that the fusion program could rapidly implement a practical application of fusion generation when one is required.

APPENDIX D
THE DATA BASE ON MgAl_2O_4

Magnesium aluminate (spinel) is a strong, mixed-oxide ceramic with good resistance to thermal stress. The properties are dependent on the preparation method, porosity, purity, and grain size of the final product. While typical properties are available, adequate for judgment of the suitability of the spinel for magnet insulation, critical properties must be evaluated on the specific product to be used in any given application.

Typical properties of MgAl_2O_4 , measured near room temperature, are given in Tables D-1, D-2, and D-3. The thermal properties (Table D-1) and mechanical properties (Table D-2) are for bulk material, but as a first approximation may be used in evaluating the performance of spray-deposited coatings. Strength properties of spray-deposited material may be somewhat low, perhaps 10 to 20%, due to entrapped porosity. Some directionality of properties can be expected in coatings, and critical properties must be measured.

The swelling and mechanical strength have been measured for neutron irradiated spinel. The data in Table D-3 show that swelling below 1% volume increase can be expected at fluences equivalent to about 2 MW-yr/m^2 for temperatures up to 750°C . Lower swelling is anticipated in FED-R (lower fluence) service. The only available measure of strength shows a 25% increase produced by irradiation at 160°C .

Data Sources

The physical, thermal, and mechanical properties were taken from: J. F. Lynch, C. G. Ruderer, and W. H. Duckworth, 'Engineering Properties of Selected Ceramic Materials' (1966), available from the American Ceramics Society, Columbus, Ohio, and also published as AFML-TR-66-52.

Table D-1. Physical and thermal properties of MgAl_2O_4

Crystal structure	Cubic
Theoretical density	3.59 g/cm ³
Melting point	2135°C
Specific heat at 25°C	800-1000 J/kg·°C
Thermal conductivity at 25°C	15 W/m·°C
Thermal expansion coefficient, 20-1100°C	$9 \times 10^{-6} \text{°C}^{-1}$

Table D-2. Mechanical properties of MgAl_2O_4 at room temperature

Bend strength	83-220 MPa
Compressive strength	1,860 MPa
Tensile strength	130 MPa
Shear strength	62 MPa
Young's modulus	234-290 GPa
Shear modulus	90-110 GPa
Bulk modulus	220 GPa
Poisson's ratio	0.29

Table D-5. The effects of neutron irradiation
on the properties of MgAl_2O_4

Irradiation temperature (°C)	Fluence ^a ($\times 10^{26}$ n/m ² , >0.1 MeV)	Displacement damage (dpa)	Swelling (%)	Strength ^b (MPa)
Control	0	0		127±4
160	2.3	30	0.8	152±11
650	2.3	23	0.2	
740	0.3	3	0.4	
825	2.3	23	1.6	

^a Irradiation in HFIR at 160°C and in EBR-II at other temperatures.

^b Strength is the tensile stress at failure for a diametral compression test of rod samples.

The properties of irradiated material (Table D-3) are from:

F. W. Clinard, Jr., L. W. Hobbs, and G. F. Hurley, "Neutron Irradiation Damage in MgO , Al_2O_3 and MgAl_2O_4 Ceramics," to be published in J. Nucl. Mater. (Proceedings of Nov. 1981 Conference at ANL on Neutron Irradiation Effects).

G. F. Jurley, J. C. Kennedy, F. W. Clinard, Jr., R. A. Youngman, and W. R. McDonnell, "Structural Properties of MgO and MgAl_2O_4 after Fission Neutron Irradiation Near Room Temperature," J. Nucl. Mater. 103-104 (1981) 761-65.

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