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UTILIZING RESISTIVE MAGNETS

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FED-R: A FUSION ENGINEERING DEVICE
UTILIZING RESISTIVE MAGNETS†

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ABSTRACT

The principal purpose of the FED-R tokamak facility is to provide a substantial quasisteady flux of fusion neutrons irradiating a large test area in order to carry out thermal, neutronic, and radiation effects testing of experimental blanket assemblies. 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, with the important exception that FED-R requires high-duty-factor operation. The outboard nuclear test region is at least 80-cm deep with approximately 60 m² of exposure area. The neutron wall loading is 0.4 MW/m² in Stage I operation ($Q_p=1.5$) and 1.3 MW/m² in Stage II ($Q_p=2.5$). The toroidal field coils are fabricated of water-cooled copper plates with demountable joints and operate steady state with a power dissipation of 180 MW in Stage I and 200 MW in Stage II.

† Presented at the Fifth Topical Meeting on the Technology of Fusion Energy, Knoxville, Tennessee, 26-28 April 1983.

I. MISSION AND OBJECTIVES OF FED-R

The primary mission of FED-R is to provide a reliable quasisteady 14-MeV neutron source of the order of 100 MW intensity to irradiate at least several tens of square meters of blanket module assemblies for the purpose of developing fusion-neutron applications such as tritium and fissile breeding, electrical power production, and radiation effects testing.¹ This mission leads to a design approach that features normal-conducting TF (toroidal-field coils), heavily driven (low- Q_p) plasma operation, and development of a first wall that is nearly "transparent" to fast neutrons.

Many different types of nuclear tests can be carried out simultaneously in a magnetic confinement fusion device. Figure 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 performance goals of FED-R are the following:

- (1) Fusion neutron wall loading $> 0.4 \text{ MW/m}^2$ over at least a 50 m^2 test area.
- (2) Fusion energy gain $Q_p > 1.0$.
- (3) Accessibility for easy change-out of nuclear test modules, loops and other assemblies, of damaged reactor components, and for maintenance.
- (4) Approximately 10^7 burn-s/yr, requiring at least 30% availability.

- (5) Steady-state operation, to be achieved with neutral-beam-driven plasma current.
- (6) Radiation level of 5 mrem/h or less outside of the reactor 36 h after shutdown.

A two-staged implementation of FED-R is proposed in order to reduce the initial capital investment in tokamak and facility subsystems. 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. This paper gives a brief summary of the FED-R design study. A detailed description can be found in Ref. 1.

II. DESIGN OVERVIEW

Figures 2 and 3 present the FED-R configuration in elevation and plan views for Stage I operation. The principal configurational difference for Stage II is that the three 150-keV positive-ion based beam injectors are to be replaced by four 250-keV negative-ion-based injectors (Ref. 1, Ch. 5). If FED-R Stage I were to be implemented at a somewhat later date than envisaged in this study, probably Stage I would use negative-ion-based injectors as well. Figure 4 shows the radial buildup of the tokamak device for both stages.

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A. Magnetic system

The FED-R magnetic system consists of 12 essentially rectangular TF coils, six EF coils, and a central OH solenoid. The TF coils, OH solenoid, and 4 internal PF coils are fabricated of water-cooled copper, while the two external PF ring coils are superconducting. The OH coil, which is used only during start-up of the plasma current, operates at B up to 9.5 T and employs a Bitter-plate winding.

B. Resistive toroidal field coils

Demountable TF coils permit the objective of having good accessibility to the in-bore components to be compatible with the cost-driving need for compactness.^{2,3} 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.

The TF coils are designed to develop 7 T and 8.8 T peak field at the coil windings for Stage I and II, respectively. By making the coils sufficiently massive, they can be operated steady state at the prescribed fields with resistive power loss of about 180 MW in Stage I and 280 MW in Stage II, or about 45% of the total facility power requirement.

The TF coils are constructed from 6-cm-thick copper plate segments that can be factory-built and shipped to the reactor site for assembly. The plates are insulated from each other by magnesium aluminate (SPINEL) applied by plasma arc spraying; this insulation can withstand at least 10^{12} rads. Out-of-plane loads are reacted by the TF coil case and intercoil support structure. The demountable finger joints incorporated into the upper and lower segments of the TF coils enhance access for replacement of the in-bore tokamak components and permit the installation of outsized nuclear test assemblies. Details of the coil design are given in Refs. 1 and 4.

C. Plasma engineering considerations

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 principal plasma parameters (T_e , T_i , β , $n\tau_E$) specified for Stage I operation are related to those expected in TFTR⁵ or in Doublet-III Upgrade. Table 1 gives a comparison between the FED-R and TFTR parameters. The vital difference in plasma operation is that TFTR has insignificant duty factor, while FED-R demands high-duty-factor operation using a single-null poloidal divertor for continuous particle control and heat removal. The separatrix created by the divertor may also be essential for achieving large values of $n\tau_E$ under conditions of high injected power.⁶

Neutral-beam injection (NBI) has been chosen as the primary plasma heating system for FED-R because of its demonstrated success in producing high-temperature, moderate-beta plasmas, excellent theoretical prospects for generating current drive (at high beam energies), and because the large increase in fusion reactivity accompanying the injected fast-ion population gives greater assurance of reaching $Q_p \sim 1$ to 3. The large injected power needed for steady-state current drive (40 to 50 MW in Stage I) is compatible with the relatively low values of Q_p specified for FED-R. Details are given in Ref. 1, Sec. 5.

Start-up of the plasma current is accomplished by a combination of 2 MW of ECRH, the OH solenoid, part of the flux swing associated with the EF system, and NB injection.

D. Neutronic considerations for the vacuum vessel

The nuclear test regions in FED-R are located outside the vacuum vessel for a number of reasons: to avoid the deleterious impact of coolant leakage from test assemblies; the need for rapid removal of test assemblies; and the unusual geometry of some test systems that prevent their introduction into the vessel.¹ Thus the first wall is chosen as the vacuum boundary. (The bulk shielding is also located completely outside the vacuum boundary.)

A number of test applications require that the outboard first wall be "neutronically thin," that is, introduce only minor attenuation of the incident fast neutron current and spectrum. These applications include blanket modules in which $(n,2n)$ and ${}^7\text{Li}(n,n')$ reactions are important, certain radiation effects programs, and the production of regions of very high temperature.

Figure 5 shows the vacuum vessel configuration. A design concept was developed for an outboard stainless steel double wall with 20-atm 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/cm^2 . The inboard, top, and bottom walls are water-cooled. Details are given in Ref. 7 and in Ref. 1, Ch. 6.

E. Nuclear testing region

Some 60 m^2 of first-wall area with at least an 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. Testing of many large-volume (1) blanket modules, (2) liquid metal loops, (3) flowing pebble beds, and (4) vertical rod assemblies (see Fig. 6) would be carried out

simultaneously with (5) radiation effects testing and (6) the production and utilization of large-cross-section fast neutron beams.

III. MAINTENANCE CONCEPT

Wherever practical, subsystems and components were designed for replacement as individual modules and for minimal impact on adjacent subsystems. This requirement is particularly important for the nuclear test modules, the divertor/pumping modules, and the vacuum vessel sectors. The test modules can be installed and removed without disturbing the vacuum system. The poloidal divertor (Fig. 5) is divided into 12 modules, each of which can be removed by radial extraction without disturbing the TF coils. The vacuum vessel is divided into 6 toroidal sectors. Replacement of a sector requires cutting the vessel sector joints and the prior removal of the outboard legs of 3 adjacent TF coils. The internal PF coils are segmented in halves, and their replacement requires the removal of the outboard legs of 4 adjacent TF coils.

The in-bore shielding is adequate to permit hands-on access to the demountable TF coil joints. Removal of the outboard leg of a TF coil requires 13 major disassembly operations, which are depicted in Fig. 7. Removal and replacement of one leg was estimated to require 350 hours.¹ Several coil legs can be replaced during the same period if a number of maintenance crews are available.

IV. COST AND POWER REQUIREMENTS

The total direct cost of Stage I is estimated to be \$810 million in 1982 dollars. The incremental cost required to upgrade FTR-R to the Stage II configuration is about \$220 million, for a total direct cost of \$1030

million. Inclusion of indirect costs (engineering, management, installation, and contingency) will increase the total cost in each case by 60% to 100%.

The total electrical power requirement for the FED-R facility is 425 MW in Stage I and 600 MW in Stage II when operating at maximum design field and injected beam power. Assuming a capacity factor of 30% averaged over the year, and an electricity cost of 30 mills/kWh, the electricity charge will be \$33 million per year in Stage I and \$49 million in Stage II, or about 25% of the annual operating cost of the facility.

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Table 1 Comparison of Principal Parameters of TFTR and FED-R

	TFTR Mid-1980s	FED-R ^a Early 1990s
Major Radius (m)	2.5	3.5
Minor Radius (m)	0.85	0.85 x 1.35
Max. B at Coil (T)	9.5	7.0 ^d 8.8 ^e
Max. B at Plasma Axis (T)	5.2	4.0 ^d 5.0 ^e
Plasma Current (MA)	3.0	3.6 ^d 4.4 ^e
N. Beam Energy (keV)	120	150 ^d
N. Beam Power (MW)	30	50 ^d
$\bar{n}_e \tau_E$ (cm ⁻³ s)	$\sim 1 \times 10^{13}$	$\sim 2 \times 10^{13}$
$\langle \beta \rangle$ = plasma pres. magnetic field pres	3 %	5 % ^b
Pulse Length (s)	~ 1	≥ 1000
Duty Factor	0.003	≥ 0.90
Fusion Gain, Q _p	~ 1	1.5 ^d
Fusion Power (MW)	20	75 ^d
Uncoll. Neutron Wall Loading (MW/m ²)	0.2	0.4 ^d (outboard)
Electrical Power Consumption (MW)	660 (short pulse)	425 ^d
Useful Irradiation Test Area (m ²)	< 4	60

^aIncluding fractional-energy neutrals.^bApprox. 1/3 of pressure is in supertherm. ions.^cNoninduct. current drive needed for $t_p > 500$ s.^dValues given are for Stage I operation.^eMaximum design values.

FIGURE CAPTIONS

- FIG. 1 Neutron source requirements for a fusion-neutron test program.
- FIG. 2 Elevation view of FED-R in Stage I configuration.
- FIG. 3 Plan view of FED-R in Stage I configuration.
- FIG. 4 Radial buildup of the FED-R device.
- FIG. 5 Cross section showing vacuum vessel configuration of FED-R.
- FIG. 6 Elevation view of illustrative vertical rod and flowing-pebble-bed nuclear test assemblies in FED-R.
- FIG. 7 Disassembly sequence for an outboard TF coil leg in FED-R.

Fig. 1 Neutron source requirements for a fusion-neutron test program.

Fig. 2 Elevation view of FED-R in Stage 1 configuration.

Fig. 3 Plan View of FED-R in Stage I Configuration.

*ALL DIMENSIONS IN cm

Fig. 4 Radial Buildup of the PED-R Device.

FED-R VACUUM VESSEL CONFIGURATION

Fig. 5 Cross section showing vacuum vessel configuration of FED-R.

Fig. 6 Elevation view of illustrative vertical rod and flowing-pebble-bed nuclear test assemblies in FED-R.

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