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PLASMA ASSESSMENTS FOR THE FUSION ENGINEERING DEVICE (FED)*

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ABSTRACT

An initial range of plasma assumptions and scenarios has been examined for the U.S. tokamak FED concept. The results suggest that the current FED baseline parameters of $R = 4.8$ m, $B_t = 3.6$ T, $a = 1.3$ m, $b = 2.1$ m (D-shape), and $I_p = 4.8$ -5.4 MA are appropriate for achieving its nominal goals of $P(\text{fusion}) = 180$ MW and a plasma $Q \gtrsim 5$ for a pulse length greater than 100 s. However, large uncertainty still exists in the areas of current startup, ion-cyclotron wave launching, influence of plasma shape on achievable beta, impurity control, plasma edge transport, and plasma disruption. Various options and remedies have been suggested to alleviate the impact of the uncertainty on the FED design concept. They appear promising because they can be studied experimentally and are not expected to lead to fundamental design modifications of FED.

1. INTRODUCTION

The U.S. Fusion Engineering Device (FED) is currently envisioned[1] to be a tokamak that produces a fusion power of 180 MW for a pulse length greater than 100 s and uses superconducting toroidal magnets with a maximum field of 8 T. It also assumes ion cyclotron resonance heating at $2\omega_{CD}$ without precluding neutral beams as an alternate, includes a pump limiter for particle exhaust with a simplified poloidal divertor as an alternate, and aims to achieve a plasma Q between 5-10 with ignition as a possibility. The baseline parameters currently selected for FED are listed in Table 1. An elevation view of the FED reference configuration is given in Fig. 1. Details of the initial FED design and trade-off studies are being made available as an ORNL technical memorandum[2].

The plasma analyses to be reported here deal with design-oriented interpretations of available physics information to guide the FED design evolution. These calculations provide basis for the baseline parameters, examine the possible range of plasma

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Table 1. Current FED baseline parameters

D-T fusion power	180 MW
Average neutral wall load	0.5 MW/m ²
ICRH power	36-50 MW
Plasma Q	≥5
Major radius, R	4.8 m
Plasma radius, a	1.3 m
Elongation, b/a	1.6
Plasma shape	DEE
Average beta, <β>	5.5%
Safety factor at edge, q _ψ	3.2
Beta poloidal/aspect ratio, εβ _p	0.5
Plasma current, I _p	4.8-5.4 MA
Maximum toroidal field, B _m	8 T
On-axis toroidal field, B _t	3.6 T
Maximum TF ripple at edge	0.8%
TF coil clear width × height	7.5 × 10.9 m
Number of full-field pulses	3.5 × 10 ⁵
Duty factor	0.6
Availability	10-20%

performance for a given baseline device and assess potentially attractive options to the baseline plasma approach. In the following we will present a description of the currently perceived scenarios of plasma operation in FED, followed by assessments of RF-assisted current startup, plasma heating, achievable

Figure 1. Elevation view of the FED reference configuration.

beta, plasma performance and operation regime, particle and impurity control options, disruption parameters, and poloidal field configuration. The paper will close with a discussion of plasma issues of large physics uncertainty, potentially of strong FED design impact, and warranting further analysis.

2. PLASMA OPERATION SCENARIO

A typical plasma discharge cycle involves prefill, ionization and electron heating, current rampup, bulk heating, burn, plasma shutdown, and pumpdown. The proposed scenario of the plasma operation through these phases is summarized in Table 2. During the ionization, electron heating and current rampup, attachment of particles to the first wall may be significant and need to be balanced by controlled gas puff. During current rampup before plasma edge reaches the pump limiter at the chamber bottom, the plasma is assumed to be in contact with a set of startup limiters near the midplane of the chamber out-board. Methods of impurity control are currently assumed to work in conjunction with the pump limiter to maintain a low plasma average charge ($Z_{eff} \leq 1.5$) with only low-Z impurities. Because of the expected high particle recycling rate ($1-5 \times 10^{23} \text{ s}^{-1}$) at the limiter, a 5-10% particle exhaust rate will be adequate in reducing the plasma density during fusion quench, assuming the absence of large particle desorption from the first wall.

Table 2
BASELINE OPERATION SCENARIO ASSUMED FOR THE FED PLASMA

PHASE	DURATION (s)	OPERATION	PLASMA BEHAVIOR
PREFILL		$p_0 < 3 \times 10^{-5} \text{ torr}$	
IONIZATION AND ELECTRON HEATING	0.2	P(ECRH) $\sim 1 \text{ MW}$ (80 GHz) X-MODE FROM HFS $V_e \rightarrow 25 \text{ V}$ $q_0 \rightarrow 3.2$	$\langle n_e \rangle \sim 10^{19} \text{ m}^{-3}$ $\langle T_e \rangle \sim 100 \text{ eV}$ NEAR $R = 5.9 \text{ m}$ (UHR) $I_p \rightarrow 0.05 \text{ MA}$, $a = 0.2 \text{ m}$
CURRENT RAMP-UP	6	$V_e = 25 \text{ V} \rightarrow 2 \text{ V}$ P(ICRH) $\rightarrow 5 \text{ MW}$ $q_0 = 3.2$	$\langle n \rangle \rightarrow 3 \times 10^{19} \text{ m}^{-3}$ $\langle T \rangle \rightarrow 1 \text{ keV}$ $I_p \rightarrow 4.8 \text{ MA}$, $a \rightarrow 1.3 \text{ m}$ $R \rightarrow 4.8 \text{ m}$, $b/a \rightarrow 1.6$
BULK HEATING	6	P(ICRH) $\leq 50 \text{ MW}$ $q_0 = 3.2$ $\epsilon \beta_p \rightarrow 0.5$ PUMP LIMITER, FUELING (IMPURITY CONTROL)	$\langle n \rangle \rightarrow 8 \times 10^{19} \text{ m}^{-3}$ $\langle T \rangle \rightarrow 10 \text{ keV}$ P(FUSION) $\rightarrow 180 \text{ MW}$ $I_p \rightarrow 5.4 \text{ MA}$ (FCT) $\langle \beta \rangle \rightarrow 5.5\%$
BURN	> 100	P(ICRH) = 36 MW	P(FUSION) = 180 MW $Z_{eff} \leq 1.5$
SHUTDOWN	10	FUSION QUENCH PARTICLE EXHAUST CURRENT QUENCH $q = 3.2$	$\langle n \rangle \rightarrow 2 \times 10^{19} \text{ m}^{-3}$ $\langle T \rangle \rightarrow 1 \text{ keV}$ $I_p \rightarrow 0.1 \text{ MA}$ $R \rightarrow 5.9 \text{ m}$
PUMPDOWN	30	$3 \times 10^{-4} \text{ torr} \rightarrow 3 \times 10^{-5} \text{ torr}$	

3. RF-ASSISTED CURRENT STARTUP

Auxiliary rf heating of electrons before and during the current rise phase in tokamak reactors has been suggested[3] to reduce the initiation loop voltage and the resistive flux expenditure, resulting in large engineering benefits. Here an extraordinary wave at 1 MW level is injected from the high field side to ionize the neutrals and heat the electrons near the upper hybrid resonance (UHR) for 0.2 s, followed by ICRH at about 5 MW level. A typical set of results[4] is given in Fig. 2, showing that in the absence of impurity, the initial electron temperature in the UHR region would be about 100 eV, permitting an initiation loop voltage of about 16 V. It is also seen that the resistive flux dissipation ($\Delta\Phi_{RES}$) could be limited to less than 10 Wb through the current rise phase. The suggested design loop voltage (25 V) has also been estimated to be tolerant to an initial oxygen content of about 1%.

Figure 2. The behavior of plasma loop voltage (V_{loop}), resistive voltage (V_{RES}), ohmic power (P_{OHMIC}), plasma current (I_p), and the average electron temperature (T_e) during rf-assisted current startup in a clean plasma.

4. BULK HEATING

Recent experimental successes[5] and the anticipated technology advantages made plasma heating in the ion cyclotron range of frequencies (ICRF) an attractive means of bulk heating in FED. Estimates of power deposition via absorption and mode conversion of a fast magnetosonic wave launched from the low field side has recently become available [6]. Figure 3 shows typical results of second harmonic heating of deuterium in an equal

mixture of deuterium and tritium and a parallel wave number of $k_{\parallel} = 0.1 \text{ cm}^{-1}$. A relatively efficient heating of deuterium at low and high values of plasma densities and temperatures is indicated.

Figure 3. Typical calculated power deposition profiles of a fast magnetosonic wave incident from the low field side near the second harmonic of deuterium:

(a) $n_e = 3 \times 10^{13} \text{ cm}^{-3}$, $T_0 = 1 \text{ keV}$, 94% wave absorption per pass, and
(b) $n_e = 1.5 \times 10^{14} \text{ cm}^{-3}$, $T_0 = 10 \text{ keV}$, 100% wave absorption per pass.

The backup option of neutral beam heating was also examined. A key question for FED is the D^+ neutral beam energy required for adequate penetration in the near-tangential direction. An answer can be obtained by calculating the neutral power into the plasma required to maintain a near-thermal-steady-state[7,8] in FED for various beam energies. A set of typical results are shown in Fig. 4, indicating that the nature of plasma bulk heating is not strongly altered over the energy range of 100-200 keV as long as the density buildup scenario[9] is assumed. A beam energy of 150 keV is chosen for design margin with reduced burden of particle exhaust via the pump limiter (Fig. 1). One-dimensional transport simulations of heating to burn have resulted in a total power requirement below 50 MW.

5. BETA CONSIDERATIONS

Recent experimental observations in ISX-B[10] have suggested that the electron energy confinement time (τ_{Ee}) degrades significantly with significant values of $\epsilon\beta_p$ (up to 0.6) or large tangential beam heating power density (up to 2 MW/m^3). Since the heating power density in FED is expected to be small by an order of magnitude, a degradation of τ_{Ee} with large $\epsilon\beta_p$ tends to be a more pessimistic assumption. Assuming a relatively general form for τ_{Ee} as

$$\tau_{Ee} \propto n_e^s R^t q_{\psi}^u B_t^v \exp[-(\epsilon\beta_p/\delta)^{\gamma}],$$

Figure 4. Contours of constant neutral beam power required to maintain a near-steady-state temperature and density, for full deuterium beam energy of (a) 100 keV, (b) 150 keV, and (c) 200 keV. Positive ion beams are assumed.

it can be shown that $n_e \tau_{Ee}$ may be written as

$$n_e \tau_{Ee} \approx B_m^{4+v} q_\psi^{u-4} F(\epsilon \beta_p) H(\kappa) G(R, a), \kappa = b/a$$

where

$$F(x) = C^4 x^2 \exp[-(x/\delta)^\gamma]$$

$$H(\kappa) = (1 + \kappa^2)^{2/4}$$

$$G(R, a) = a^5 R^t \epsilon^2 (1 - \epsilon - \frac{\Delta_s}{R})^{4+u} / (1 - \epsilon^2)^8.$$

Here C ($=1.1-1.3$) is a mildly increasing function of $\epsilon \beta_p$ and q_ψ and Δ_s denotes the distance between the plasma and the inner TF coil legs.

From these expressions it is seen that the uncertainties in the size-scaling of τ_{Ee} does not influence the proper choices of $\epsilon \beta_p$ and κ , and that assuming $\kappa = 1.6$ leads to a factor-of-three enhancement in $n_e \tau_{Ee}$. Over a wide range of values for δ and γ , $F(\epsilon \beta)$ maximizes over the range of $\epsilon \beta_p = 0.5-0.6$. The assumption of $\langle \beta \rangle = 5.5\%$ for FED in Tables 1 and 2 is thus consistent with $\epsilon \beta_p = 0.5$, $q_\psi = 3.2$ and $I_p = 5.4$ MA.

6. PLASMA PERFORMANCE AND OPERATION REGIME

The WHIST one-and-one-half-dimensional (1-1/2-D) time-dependent transport code[8] has recently been modified to generate quasi-static plasma operation parameters and profiles. The information can be displayed in the form of Plasma Operation CONTOUR (POPCON) plots in the average density-temperature space[11] for the FED plasma. The physics assumptions used include: (1) injection of 150-keV neutral deuterium beams tangential to the inside edge of the plasma with a neutral power mix of 66:19:15 among full, half, and third energy components; (2) an ion conduction model that includes twice neoclassical losses plus flux surface integrated ripple-plateau[12] and ripple-trapped[13] losses for 0.7% peak-to-average ripple at the plasma edge; (3) Alcator scaling for electron conduction; (4) local fusion alpha heat deposition; (5) pellet fueling coupled with complete particle exhaust; and (6) absence of impurities.

The initial results are given in Fig. 5, indicating that the nominal FED goals of $Q \geq 5$, $P_{inj} \leq 36$ MW, $\langle \beta \rangle \leq 5.5\%$ and $P(fusion) \leq 200$ MW can be satisfied over a relatively wide regime in density ($0.45 \times 10^{14} \text{ cm}^{-3}$ to $1.2 \times 10^{14} \text{ cm}^{-3}$) and temperature (6.5 keV to 14 keV). The size of this operation regime has also been shown to permit a factor-of-two degradation in χ_e .

Figure 5. Regimes of densities and temperatures meeting the conditions of $Q > 5$, $P_{inj} \leq 36$ MW, $\langle \beta \rangle \leq 5.5\%$, and $P_{fusion} \leq 200$ MW.

7. PARTICLE AND IMPURITY CONTROL

Several approaches to impurity and particle control including baseline pump limiter[2], backup magnetic divertor[14,15], impurity flow reversal[16] and cold plasma edge[17] are being assessed for application to FED. The relative effectiveness and the major issues of these approaches are listed in Table 3.

Table 3. Relative Effectiveness and Major Issues of Various Approaches to Impurity and Particle Control

APPROACH	ESTIMATED EFFECTIVENESS		MAJOR ISSUES
	PARTICLE EXHAUST	IMPURITY CONTROL	
PUMP LIMITER (BASELINE)	ADEQUATE	UNKNOWN	<ul style="list-style-type: none"> • IMPURITY CONTAMINATION • LIMITER EROSION
POLOIDAL DIVERTOR (ALTERNATE)	ADEQUATE	ADEQUATE	<ul style="list-style-type: none"> • CONFIGURATION (INTOR) • PLATE EROSION
BUNDLE DIVERTOR (SUPPLEMENT)	MODEST	MODEST	<ul style="list-style-type: none"> • IMPACT OF FIELD RIPPLE
IMPURITY FLOW REVERSAL	—	UNCERTAIN	<ul style="list-style-type: none"> • PLASMA EDGE AND IMPURITY TRANSPORT
COLD EDGE (E.G. PLASMA FLUSH)	—	UNCERTAIN	<ul style="list-style-type: none"> • RADIATIVE COLLAPSE • PARTICLE EXHAUST

The pump limiter (Fig. 1) is in the form of a toroidal belt and is composed of ten equal segments, each retrievable between the TF coils. While adequate particle exhaust (5-10% of particle flux to limiter) is anticipated, impurity control methods such as those listed in Table 3 may be needed as a supplement. The poloidal divertor concept as an alternate to the pump limiter will borrow heavily from the INTOR results[14], with the possible exception of using copper coils interior to the TF coils.

Preliminary estimates of the plasma particle and power loads to the limiter have recently been obtained using the ORNL PROCTR transport code[18] for ICRH and NB heating cases. As summarized in Table 4, uncertainty in the level of impurity contamination leads to order-of-magnitude variations in the power to the limiter and a factor of 3-4 variations in the lifetime of the limiter. In the case of ICRH, the need of large source at the plasma edge can lead to high edge density and particle flux, resulting in unacceptably large limiter erosion and short lifetime. The central fueling by NB (at a rate of about $2 \times 10^{21} \text{ s}^{-1}$) can lead to the inverse results. Although large uncertainty exists in this physics area, the basic FED design apparently can accommodate all these options.

8. PLASMA DISRUPTION ASSUMPTIONS

A two-phase disruption model is assumed for the current FED design studies. As depicted in Fig. 6 a disruption is preceded by tearing mode activities[19] which lead to a sudden and large enhancement of plasma heat loss (thermal quench) followed by a loss of plasma current and position (current quench).

Table 4. Preliminary Estimates of the Plasma Particle and Power Loads to the Pump Limiter Under the Assumptions of a Clean or Mildly Contaminated Plasma, Using ICRH or NB Heating

	CLEAN		MILDLY CONTAMINATED	
	ICRH	NB	ICRH	NB
POWER TO LIMITER (MW)	50	50	5	5
ION FLUX TO LIMITER (s^{-1})	5×10^{23}	5×10^{22}	5×10^{23}	5×10^{22}
n_{edge} (cm^{-3})	6.5×10^{13}	3.7×10^{12}	1.2×10^{14}	6.5×10^{12}
T_{edge} (eV)	160	1600	16	160
λ_n (cm)	3.2	1.8	5.7	3.2
λ_Q (cm)	1.6	0.9	2.9	1.6
LIMITER LIFE-TIME (M) (20% AVAILABILITY)	2.6	70	9.5	26

A set of tentative plasma disruption parameters is summarized in Table 5, together with additional assumptions and comments. Care has been taken to preserve some pessimism in the assumptions. Initial design assessments[2] have indicated that the graphite armor can survive hundreds of full power disruptions. These parameters and descriptions will be updated as additional information becomes available.

9. POLOIDAL FIELD CONFIGURATIONS

A relatively simple poloidal field coil configuration has been obtained for FED. As shown in Fig. 7a, it consists of an ohmic heating solenoid (superconducting) with a separately controlled inner segment; two D-shaping coil bundles (copper) interior to the TF coils; and two outer coils (superconducting) that provide a large portion of the vertical field. A sequence of MHD equilibrium calculations has been obtained to verify that the coil locations are appropriate in producing the initial field null at $t = 0$ s (Fig. 7b), and properly positioning and shaping the plasma through the discharge cycle (Figs. 7c-7f). Aside from the solenoid, the total current in the coils is less than 20 mega-ampere-turns (MAT). Each coil bundle in this concept is required to provide the equilibrium field as well as the current drive functions, permitting a highly simplified configuration.

Figure 6. Schematic plasma behavior during a typical major disruption. The disruption is currently assumed to be preceded by increased activities of $m = 2/n = 1$ and other tearing modes in a timescale of $\Delta t_{2/l}$, to proceed with a thermal quench in a timescale of Δt_T , and to terminate with a current quench and loss of plasma position in Δt_I .

10. DISCUSSION

The results discussed above serve to highlight the initial range of plasma assumptions and scenarios examined in the current design assessments of the tokamak FED. They suggest that the current FED baseline parameters of $R = 4.8$ m, $B_t = 3.6$ T, $a = 1.3$ m, $b = 2.1$ m (D-shape), and $i_p = 4.8$ -5.4 MA are appropriate for achieving the FED design goals of $P(\text{fusion}) \approx 180$ MW and a plasma $Q \geq 5$.

While we are continuing to refine our estimates of the FED plasma behavior and performance, several issues stand out because of the uncertainty in physics and their potentially strong impact on the FED design. These areas include current startup, ion-cyclotron wave launching, influence of plasma shape on the achievable beta, impurity control, plasma edge transport, and plasma disruption. They are emphasized in our continuing effort of FED design studies.

Figure 7. (a) Poloidal field coil configuration and the poloidal flux plots at (b) $t = 0$ s, (c) $t = 2$ s, (d) $t = 6$ s, (e) $t = 12$ s and (f) $t = 112$ s.

Table 5. Working Plasma Disruption Parameters for FED

Parameter	Value	Assumptions/comments
Thermal quench phase		
Time scale, t_T	5 ms	About 5% of τ_{Fe} under normal conditions
Thermal energy deposition	90 MJ	Concomitant fusion quench and stopping supplementary heating
Deposition region		
Limiter (via particle)	45 MJ	Little change in plasma position and shape
First wall (via radiation)	45 MJ	Due to ablated impurities
Voltage spike	1 kV	
Plasma flux reduction	5 Wb	
Current quench phase		
Time scale, t_I	10 ms	Short compared with the first wall eddy current decay time
Thermal energy deposition	10 MJ	Ohmic plasma energy
Region of plasma impact		Inboard, top, or bottom
Affected area/chamber wall	10%	Uncertain
Energy deposition peaking factor	10	To retain some pessimism
Plasma magnetic energy	60 MJ	Dissipated via eddy currents in nearby conductors

Various options and remedies have been suggested and assessed to alleviate the impact of the uncertainty on the FED design concept. They appear promising because they can be studied experimentally and are not expected to lead to fundamental design modifications of FED.

REFERENCES

- [1] These major features were recently approved by the Technical Management Board (TMB) of the U.S. Magnetic Fusion Program. The TMB is currently composed of R. Conn (Chairman), J. Gilleland (Executive Director), H. Furth, B. Montgomery, J. Sheffield, K. Thomassen and L. Ledman (Executive Secretary).
- [2] FLANAGAN, C. A., STEINER, D. and SMITH, G. (Editors), "Initial Trade and Design Studies for the Fusion Engineering Device," ORNL/TM-7777 (to be published).

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Plasma shape	DEE
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Safety factor at edge, q_ψ	3.2
Beta poloidal/aspect ratio, $\epsilon\beta_p$	0.5
Plasma current, I_p	4.8-5.4 MA
Maximum toroidal field, B_m	8 T
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CURRENT RAMP-UP	6	$V_E = 25 \text{ V} \rightarrow 2 \text{ V}$ P(ICRH) $\sim 5 \text{ MW}$ $q_0 = 3.2$	$(n) \rightarrow 3 \times 10^{19} \text{ m}^{-3}$ $(T) \rightarrow 1 \text{ keV}$ $I_p \rightarrow 4.8 \text{ MA}$, a $\sim 1.3 \text{ m}$ R $\rightarrow 4.8 \text{ m}$, b/a ~ 1.6
BULK HEATING	6	P(ICRH) $\leq 50 \text{ MW}$ $q_0 = 3.2$ $\epsilon\beta_p \rightarrow 0.5$ PUMP LIMITER, FUELING (IMPURITY CONTROL)	$(n) \rightarrow 8 \times 10^{19} \text{ m}^{-3}$ $(T) \rightarrow 10 \text{ keV}$ P(FUSION) $\sim 180 \text{ MW}$ $I_p \sim 5.4 \text{ MA}$ (FCT) $\langle\beta\rangle \sim 5.5\%$
BURN	> 100	P(ICRH) = 36 MW	P(FUSION) = 180 MW $Z_{eff} \leq 1.5$
SHUTDOWN	10	FUSION QUENCH PARTICLE EXHAUST CURRENT QUENCH $q = 3.2$	$(n) \rightarrow 2 \times 10^{19} \text{ m}^{-3}$ $(T) \rightarrow 1 \text{ keV}$ $I_p \rightarrow 0.1 \text{ MA}$ R $\rightarrow 5.9 \text{ m}$
PUMPDOWN	30	$3 \times 10^{-4} \text{ torr} \rightarrow 3 \times 10^{-5} \text{ torr}$	

3. RF-ASSISTED CURRENT STARTUP

Auxiliary rf heating of electrons before and during the current rise phase in tokamak reactors has been suggested[3] to reduce the initiation loop voltage and the resistive flux expenditure, resulting in large engineering benefits. Here an extraordinary wave at 1 MW level is injected from the high field side to ionize the neutrals and heat the electrons near the upper hybrid resonance (UHR) for 0.2 s, followed by ICRH at about 5 MW level. A typical set of results[4] is given in Fig. 2, showing that in the absence of impurity, the initial electron temperature in the UHR region would be about 100 eV, permitting an initiation loop voltage of about 16 V. It is also seen that the resistive flux dissipation ($\Delta\phi_{RES}$) could be limited to less than 10 Wb through the current rise phase. The suggested design loop voltage (25 V) has also been estimated to be tolerant to an initial oxygen content of about 1%.

Figure 2. The behavior of plasma loop voltage (V_{loop}), resistive voltage (V_{RES}), ohmic power (P_{OHMIC}), plasma current (I_p), and the average electron temperature (T_e) during rf-assisted current startup in a clean plasma.

4. BULK HEATING

Recent experimental successes[5] and the anticipated technology advantages made plasma heating in the ion cyclotron range of frequencies (ICRF) an attractive means of bulk heating in FED. Estimates of power deposition via absorption and mode conversion of a fast magnetosonic wave launched from the low field side has recently become available [6]. Figure 3 shows typical results of second harmonic heating of deuterium in an equal

mixture of deuterium and tritium and a parallel wave number of $k_{\parallel} = 0.1 \text{ cm}^{-1}$. A relatively efficient heating of deuterium at low and high values of plasma densities and temperatures is indicated.

Figure 3. Typical calculated power deposition profiles of a fast magnetosonic wave incident from the low field side near the second harmonic of deuterium:

(a) $n_e = 3 \times 10^{13} \text{ cm}^{-3}$, $T_0 = 1 \text{ keV}$, 94% wave absorption per pass, and
 (b) $n_e = 1.5 \times 10^{14} \text{ cm}^{-3}$, $T_0 = 10 \text{ keV}$, 100% wave absorption per pass.

The backup option of neutral beam heating was also examined. A key question for FED is the D^+ neutral beam energy required for adequate penetration in the near-tangential direction. An answer can be obtained by calculating the neutral power into the plasma required to maintain a near-thermal-steady-state[7,8] in FED for various beam energies. A set of typical results are shown in Fig. 4, indicating that the nature of plasma bulk heating is not strongly altered over the energy range of 100-200 keV as long as the density buildup scenario[9] is assumed. A beam energy of 150 keV is chosen for design margin with reduced burden of particle exhaust via the pump limiter (Fig. 1). One-dimensional transport simulations of heating to burn have resulted in a total power requirement below 50 MW.

5. BETA CONSIDERATIONS

Recent experimental observations in ISX-B[10] have suggested that the electron energy confinement time (τ_{Ee}) degrades significantly with significant values of $\epsilon\beta_p$ (up to 0.6) or large tangential beam heating power density (up to 2 MW/m^3). Since the heating power density in FED is expected to be small by an order of magnitude, a degradation of τ_{Ee} with large $\epsilon\beta_p$ tends to be a more pessimistic assumption. Assuming a relatively general form for τ_{Ee} as

$$\tau_{Ee} \propto n_e^s a^s R^t q_{\psi}^u B_t^v \exp[-(\epsilon\beta_p/\delta)^Y],$$

Figure 4. Contours of constant neutral beam power required to maintain a near-steady-state temperature and density, for full deuterium beam energy of (a) 100 keV, (b) 150 keV, and (c) 200 keV. Positive ion beams are assumed.

it can be shown that $n_e \tau_{Ee}$ may be written as

$$n_e \tau_{Ee} \propto B_m^{4+v} q_\psi^{-4} F(\epsilon \beta_p) H(\kappa) G(R, a), \kappa = b/a$$

where

$$F(x) = C^4 x^2 \exp[-(x/\delta)^\gamma]$$

$$H(\kappa) = (1 + \kappa^2)^{2/4}$$

$$G(R, a) = a^5 R^t \epsilon^2 (1 - \epsilon - \frac{\Delta_s}{R})^{4+u} / (1 - \epsilon^2)^8.$$

Here C ($=1.1-1.3$) is a mildly increasing function of $\epsilon \beta_p$ and q_ψ and Δ_s denotes the distance between the plasma and the inner TF coil legs.

From these expressions it is seen that the uncertainties in the size-scaling of τ_{Ee} does not influence the proper choices of $\epsilon \beta_p$ and κ , and that assuming $\kappa = 1.6$ leads to a factor-of-three enhancement in $n_e \tau_{Ee}$. Over a wide range of values for δ and γ , $F(\epsilon \beta)$ maximizes over the range of $\epsilon \beta_p = 0.5-0.6$. The assumption of $\langle \beta \rangle = 5.5\%$ for FED in Tables 1 and 2 is thus consistent with $\epsilon \beta_p = 0.5$, $q_\psi = 3.2$ and $I_p = 5.4$ MA.

6. PLASMA PERFORMANCE AND OPERATION REGIME

The WHIST one-and-one-half-dimensional (1-1/2-D) time-dependent transport code[8] has recently been modified to generate quasi-static plasma operation parameters and profiles. The information can be displayed in the form of Plasma Operation CONtour (POPCON) plots in the average density-temperature space[11] for the FED plasma. The physics assumptions used include: (1) injection of 150-keV neutral deuterium beams tangential to the inside edge of the plasma with a neutral power mix of 66:19:15 among full, half, and third energy components; (2) an ion conduction model that includes twice neoclassical losses plus flux surface integrated ripple-plateau[12] and ripple-trapped[13] losses for 0.7% peak-to-average ripple at the plasma edge; (3) Alcator scaling for electron conduction; (4) local fusion alpha heat deposition; (5) pellet fueling coupled with complete particle exhaust; and (6) absence of impurities.

The initial results are given in Fig. 5, indicating that the nominal FED goals of $Q \geq 5$, $P_{inj} \leq 36$ MW, $\langle \beta \rangle \leq 5.5\%$ and $P(fusion) \leq 200$ MW can be satisfied over a relatively wide regime in density ($0.45 \times 10^{14} \text{ cm}^{-3}$ to $1.2 \times 10^{14} \text{ cm}^{-3}$) and temperature (6.5 keV to 14 keV). The size of this operation regime has also been shown to permit a factor-of-two degradation in χ_e .

Figure 5. Regimes of densities and temperatures meeting the conditions of $Q > 5$, $P_{inj} \leq 36$ MW, $\langle \beta \rangle \leq 5.5\%$, and $P_{fusion} \leq 200$ MW.

7. PARTICLE AND IMPURITY CONTROL

Several approaches to impurity and particle control including baseline pump limiter[2], backup magnetic divertor[14,15], impurity flow reversal[16] and cold plasma edge[17] are being assessed for application to FED. The relative effectiveness and the major issues of these approaches are listed in Table 3.

Table 3. Relative Effectiveness and Major Issues of Various Approaches to Impurity and Particle Control

APPROACH	ESTIMATED EFFECTIVENESS		MAJOR ISSUES
	PARTICLE EXHAUST	IMPURITY CONTROL	
PUMP LIMITER (BASELINE)	ADEQUATE	UNKNOWN	<ul style="list-style-type: none"> • IMPURITY CONTAMINATION • LIMITER EROSION
POLOIDAL DIVERTOR (ALTERNATE)	ADEQUATE	ADEQUATE	<ul style="list-style-type: none"> • CONFIGURATION (INTOR) • PLATE EROSION
BUNDLE DIVERTOR (SUPPLEMENT)	MODEST	MODEST	<ul style="list-style-type: none"> • IMPACT OF FIELD RIPPLE
IMPURITY FLOW REVERSAL	—	UNCERTAIN	<ul style="list-style-type: none"> • PLASMA EDGE AND IMPURITY TRANSPORT
COLD EDGE (E.G. PLASMA FLUSH)	—	UNCERTAIN	<ul style="list-style-type: none"> • RADIATIVE COLLAPSE • PARTICLE EXHAUST

The pump limiter (Fig. 1) is in the form of a toroidal belt and is composed of ten equal segments, each retrievable between the TF coils. While adequate particle exhaust (5-10% of particle flux to limiter) is anticipated, impurity control methods such as those listed in Table 3 may be needed as a supplement. The poloidal divertor concept as an alternate to the pump limiter will borrow heavily from the INTOR results[14], with the possible exception of using copper coils interior to the TF coils.

Preliminary estimates of the plasma particle and power loads to the limiter have recently been obtained using the ORNL PROCTR transport code[18] for ICRH and NB heating cases. As summarized in Table 4, uncertainty in the level of impurity contamination leads to order-of-magnitude variations in the power to the limiter and a factor of 3-4 variations in the lifetime of the limiter. In the case of ICRH, the need of large source at the plasma edge can lead to high edge density and particle flux, resulting in unacceptably large limiter erosion and short lifetime. The central fueling by NB (at a rate of about $2 \times 10^{21} \text{ s}^{-1}$) can lead to the inverse results. Although large uncertainty exists in this physics area, the basic FED design apparently can accommodate all these options.

8. PLASMA DISRUPTION ASSUMPTIONS

A two-phase disruption model is assumed for the current FED design studies. As depicted in Fig. 6 a disruption is preceded by tearing mode activities[19] which lead to a sudden and large enhancement of plasma heat loss (thermal quench) followed by a loss of plasma current and position (current quench).

Table 4. Preliminary Estimates of the Plasma Particle and Power Loads to the Pump Limiter Under the Assumptions of a Clean or Mildly Contaminated Plasma, Using ICRH or NB Heating

	CLEAN		MILDLY CONTAMINATED	
	ICRH	NB	ICRH	NB
POWER TO LIMITER (MW)	50	50	5	5
ION FLUX TO LIMITER (s^{-1})	5×10^{23}	5×10^{22}	5×10^{23}	5×10^{22}
n_{edge} (cm^{-3})	6.5×10^{13}	3.7×10^{12}	1.2×10^{14}	6.5×10^{12}
T_{edge} (eV)	160	1600	16	160
λ_n (cm)	3.2	1.8	5.7	3.2
λ_Q (cm)	1.6	0.9	2.9	1.6
LIMITER LIFE-TIME (M) (20% AVAILABILITY)	2.6	70	9.5	26

A set of tentative plasma disruption parameters is summarized in Table 5, together with additional assumptions and comments. Care has been taken to preserve some pessimism in the assumptions. Initial design assessments[2] have indicated that the graphite armor can survive hundreds of full power disruptions. These parameters and descriptions will be updated as additional information becomes available.

9. POLOIDAL FIELD CONFIGURATIONS

A relatively simple poloidal field coil configuration has been obtained for FED. As shown in Fig. 7a, it consists of an ohmic heating solenoid (superconducting) with a separately controlled inner segment; two D-shaping coil bundles (copper) interior to the TF coils; and two outer coils (superconducting) that provide a large portion of the vertical field. A sequence of MHD equilibrium calculations has been obtained to verify that the coil locations are appropriate in producing the initial field null at $t = 0$ s (Fig. 7b), and properly positioning and shaping the plasma through the discharge cycle (Figs. 7c-7f). Aside from the solenoid, the total current in the coils is less than 20 mega-ampere-turns (MAT). Each coil bundle in this concept is required to provide the equilibrium field as well as the current drive functions, permitting a highly simplified configuration.

Figure 6. Schematic plasma behavior during a typical major disruption. The disruption is currently assumed to be preceded by increased activities of $m = 2/n = 1$ and other tearing modes in a timescale of $\Delta t_{2/1}$, to proceed with a thermal quench in a timescale of Δt_T , and to terminate with a current quench and loss of plasma position in Δt .

10. DISCUSSION

The results discussed above serve to highlight the initial range of plasma assumptions and scenarios examined in the current design assessments of the tokamak FED. They suggest that the current FED baseline parameters of $R = 4.8$ m, $B_t = 3.6$ T, $a = 1.3$ m, $b = 2.1$ m (D-shape), and $I_p = 4.8$ -5.4 MA are appropriate for achieving the FED design goals of $P(\text{fusion}) \approx 180$ MW and a plasma $Q \geq 5$.

While we are continuing to refine our estimates of the FED plasma behavior and performance, several issues stand out because of the uncertainty in physics and their potentially strong impact on the FED design. These areas include current startup, ion-cyclotron wave launching, influence of plasma shape on the achievable beta, impurity control, plasma edge transport, and plasma disruption. They are emphasized in our continuing effort of FED design studies.

Figure 7. (a) Poloidal field coil configuration and the poloidal flux plots at (b) $t = 0$ s, (c) $t = 2$ s, (d) $t = 6$ s, (e) $t = 12$ s and (f) $t = 112$ s.

Table 5. Working Plasma Disruption Parameters for FED

Parameter	Value	Assumptions/comments
Thermal quench phase		
Time scale, t_T	5 ms	About 5% of τ_{Fe} under normal conditions
Thermal energy deposition	90 MJ	Concomitant fusion quench and stopping supplementary heating
Deposition region		
Limiter (via particle)	45 MJ	Little change in plasma position and shape Due to ablated impurities
First wall (via radiation)	45 MJ	
Voltage spike	1 kV	
Plasma flux reduction	5 Wb	
Current quench phase		
Time scale, t_I	10 ms	Short compared with the first wall eddy current decay time
Thermal energy deposition	10 MJ	Ohmic plasma energy
Region of plasma impact		Inboard, top, or bottom
Affected area/chamber wall	10%	Uncertain
Energy deposition peaking factor	10	To retain some pessimism
Plasma magnetic energy	60 MJ	Dissipated via eddy currents in nearby conductors

Various options and remedies have been suggested and assessed to alleviate the impact of the uncertainty on the FED design concept. They appear promising because they can be studied experimentally and are not expected to lead to fundamental design modifications of FED.

REFERENCES

- [1] These major features were recently approved by the Technical Management Board (TMB) of the U.S. Magnetic Fusion Program. The TMB is currently composed of R. Conn (Chairman), J. Gilleland (Executive Director), H. Furth, B. Montgomery, J. Sheffield, K. Thomassen and L. Ledman (Executive Secretary).
- [2] FLANAGAN, C. A., STEINER, D. and SMITH, G. (Editors), "Initial Trade and Design Studies for the Fusion Engineering Device," ORNL/TM-7777 (to be published).

- [3] PENG, Y-K. M., BOROWSKI, S. K. and KAMMASH, T., Nucl. Fusion 18, 1489 (1978).
- [4] BOROWSKI, S. K., PENG, Y-K. M. and KAMMASH, T., "RF-Assisted Current Startup of Large Tokamak Plasmas," to be published.
- [5] HOSEA, J. C. et al., Paper IAEA-CN-38/D-5-1; TFR Group, Paper IAEA-CN-38/D-3; KIMURA, H. et al., Paper IAEA-CN-38/D-5-2, to be published in Plasma Physics and Controlled Nuclear Fusion Research 1980.
- [6] BLACKFIELD, D., "Numerical Simulations of ICRF Heated Tokamak Plasmas," UWFD-353, University of Wisconsin (May 1980); BLACKFIELD, D. and COHN, D. R., private communication.
- [7] WATKINS, M. L. et al., Paper IAEA-CN-38/W-3, to be published in Plasma Physics and Controlled Nuclear Fusion Research 1980.
- [8] HOULBERG, W. A. et al., "Developments in Tokamak Transport Modeling," paper presented at ANS/ENS Topical Meeting on Advances in Mathematical Methods for Nuclear Engineering Problems, Munich, Federal Republic of Germany, April 27-29, 1981.
- [9] HOLMES, J. A. et al., Nucl. Fusion 20, 59 (1980).
- [10] MURAKAMI, M. et al., Paper IAEA-CN-38/M-1, to be published in Plasma Physics and Controlled Nuclear Fusion Research 1980; SWAIN, D. W., to be published.
- [11] ATTENBERGER, S. E. and HOULBERG, W. A., "Heating Power Contours for the FED Tokamak," to be presented at the 1981 IEEE International Conference on Plasma Science, Santa Fe, New Mexico, May 18-20, 1981.
- [12] BOOZER, A., PPPL-1619; GOLDSTON, R. J. and TOWNER, H. H., PPPL-1637, Princeton Plasma Physics Laboratory (1980).
- [13] GOLDSTON, R. J. and TOWNER, H. H., Nucl. Fusion 20, 781 (1980).
- [14] SCHMIDT, J. A. et al., "INTOR Phase I Poloidal Divertor Studies," to be published.
- [15] HIVELEY, L. M. et al., "Physics Optimization of Bundle Divertors," to be published.
- [16] STACEY, W. M. and SIGMAR, D. J., Nucl. Fusion 19, 1665 (1979); BURRELL, K. H. et al., Nucl. Fusion 20, 1021 (1980).
- [17] NEUHAUSER, J. IPP-1/182, Max-Planck-Institut für Plasmaphysik (August 1980).
- [18] HOWE, H. C. et al., "PROCTR Formulary," to be published as ORNL Technical Memorandum.
- [19] CARRERAS, B. et al., Phys. Fluids 23, 1811 (1980); HICKS, H. R. et al., ORNL/TM-7132, Oak Ridge National Laboratory (1981).