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Fusion-Fission Research Facility (FFRF) as a Practical Step Toward Hybrids

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Abstract: The project of ASIPP (with PPPL participation), called FFRF, (R/a=4/1 m/m, I_{pl}=5 MA, B_{tor}=4-6 T, P_{DT}=50-100 MW, $P_{fission}$=80-4000 MW, 1 m thick blanket) is outlined. FFRF stands for the Fusion-Fission Research Facility with a unique fusion mission and a pioneering mission of merging fusion and fission for accumulation of design, experimental, and operational data for future hybrid applications. The design of FFRF will use as much as possible the EAST and ITER design experience. On the other hand, FFRF strongly relies on new, Lithium Wall Fusion plasma regimes, the development of which has already started in the US and China.

Key words: fusion power, tokamaks, fusion-fission hybrid, burning plasma

1 Mission and scientific strategy

FFRF is a project of ASIPP, which is an institution working on the development of fusion applications to nuclear energy.

The mission of FFRF is to advance fusion to the level of a stationary neutron source and to create a technical, scientific, and technology basis for the utilization of high-energy fusion neutrons for the needs of nuclear energy and technology.

The mission of FFRF is unique, ambitious, and at the same time realistic. It took more than 25 years for the ITER project [1] to approach the construction phase, and still there are numerous uncertainties in the design of its critical elements and systems. The major reason of slow progress in ITER is related to uncertainties in the plasma regime, whose understanding is still evolving without giving sufficient confidence in performance of the next step facility.

Unlike ITER, which has chosen the conventional plasma physics concept for a burning plasma regime, FFRF takes advantage of a new approach to magnetic fusion, which emerged during the last decade (since Dec. 1998) [2, 3, 4].

For magnetically confined plasma, it is much more efficient to prevent its cooling by neutrals recycled from the walls, rather than to rely, as the conventional approach, on extensive heating power in order to compensate the otherwise essentially unlimited cooling.

In the 70s, this understanding was behind several fusion reactor projects with the special magnetic configurations consisting of closed, nested magnetic surfaces, which are separated by a separatrix from the open magnetic surfaces. The plasma has to be confined in the closed magnetic surfaces. The plasma particle flux, which crosses the separatrix should be directed along the open field lines through narrow channels to the pumping volume. Plasma particle pumping was the essence of this "pumping divertor" approach.

Then, in the early 80s, after discovery on the ASDEX device of the so-called H-mode regime with enhanced confinement, the idea of a pumping divertor has been abandoned. The expectations were created that sufficient performance can be obtained with H-mode even in simple magnetic configurations, without design and operational complexities of the pumping divertor. But, in fact, the unanimous and uncritical adoption of H-mode, has deviated magnetic fusion from its basic line. The plasma physics problems, instead of being resolved, were accumulated at the larger scale in big fusion machines. Unresolved, the same problems were amplified in the ITER project, explaining its outstanding design challenges and delays.

The fundamental shift back to the basic line in magnetic fusion was made in Dec. 1998 when T-11M [5, 6] experiments have demonstrated the outstanding abilities of lithium coating in absorbing deuterium plasma particles. It was immediately understood that the lithium plasma facing layer can eliminate the recycling and prevent plasma edge cooling by the cold neutrals.

Complemented in 2006 with the core fueling by the Neutral Beam Injection (NBI), the lithium wall idea resulted in the LiWall Fusion concept of magnetic fusion, now completed. Its comprehensive theoretical analysis during the last 10 years and initial experiments on several machines around the world suggest the unambiguous superiority of the LiWF [4] over the conventional approach. The LiWF can solve the otherwise unresolvable plasma physics and fusion technology problems instead of accumulating them in the next step devices [7, 8].

The strategy of FFRF is to design the machine in parallel with the supporting experimental and technology development of the LiWF regimes.

This FFRF strategy is significantly different from the initially adopted ITER strategy with a never realized conservative reliance on "well established data and understanding" in plasma physics. In contrast, FFRF relies on development of plasma regimes, which eliminate dependence on numerous uncertainties in the tokamak plasma physics.

The mission of FFRF essentially determines the major parameters of the machine. Based on a superconducting tokamak, FFRF will be a long lasting research facility. The requirement of 1 m thick blanket for protecting super-conducting coils from the neutron radiation dictated the large size of the machine. At the same time the plasma physics requirements limit the enlargement of the machine. The compromise solution is at the major radius of about 4 m and the plasma current of about 5 MA.

Even with existing uncertainties in the plasma and blanket regimes, these basic requirements specify the major parameters of FFRF, thus, allowing the design of the time- and labor-consuming systems of the machine. The design of other systems, which are related to the details of plasma control and blanket design, can follow upon accumulation of necessary experimental information.

Being a conventional tokamak with the size between EAST [9] and ITER [10], FFRF will rely as much as possible on their existing design. Thus, the magnetic system, especially Toroidal Field Coils (TFC), can take advantage of the ITER experience. TFC in FFRF can use the same superconductor as ITER. The plasma regimes, on the other hand, will represent an extension of the stationary plasma regimes on HT-7 [11] and EAST tokamaks at ASIPP. Both pulsed inductive discharges and stationary non-inductive Lower Hybrid Current Drive (LHCD) will be possible (although only the first one is considered in the present paper).

2 Reference parameters and burning plasma regime of FFRF

The Table below specifies the reference FFRF parameters

Table 1. FFRF parameters.

Parameter	Value	Parameter	Value
$D_{blanket,m}$	1	a_m, R_m	1.0, 4.0
$V^{pl}_{m^3}, S^{pl}_{m^2}$	130, 230	$I_{pl,MA}$	5.
$B_{tor,T}$	4-6	$\Delta\Psi_{f-top,V sec}$	40
n_{20}	0.4	$\frac{T_i+T_e}{2}\|_{keV}$	24-27
E^{NBI}_{keV}	120	P^{NBI}_{MW}	2-5
$\tau^{IND}_{E,sec}$	20-7	P^{DT}_{MW}	50-100

Here, a, R are minor and major radii of the plasma, V, S are its volume and the surface area, n is the plasma density, E^{NBI} is the energy of Neutral Beam Injection, T_i, T_e are electron and ion temperatures, B_{tor}, I_{pl} are the toroidal magnetic field (at plasma geometric center) and the plasma current, $\Delta\Psi_{f-top}$ the resistive Volt-second for the flat-top of the current, W_{MJ} is the total thermal energy of the plasma, τ_E^{IND} is the energy confinement time (inductive regime), P^{NBI} is the NBI power, P^{DT} is the fusion power.

The power of the active fission core power is not yet specified but can be within 80-4000 MW, depending on the fuel composition. Its design and regimes are beyond the scope of the present paper and is covered elsewhere (see, e.g., [12]).

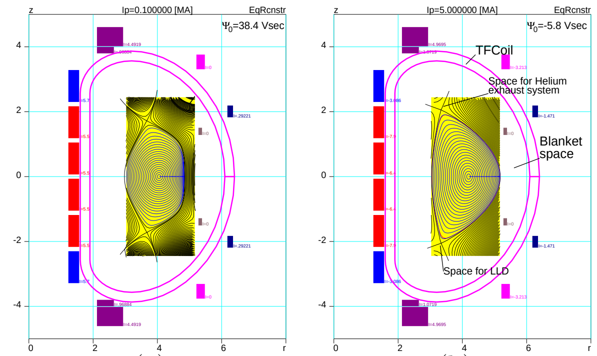
The magnetic system, based on the ITER superconductor, would be capable of the toroidal magnetic field of up to 6 T. In fact, for fusion power of 50 MW, the toroidal field $B_{tor} = 4$ T is sufficient for the LiWF regime. In parallel with progress in ITER plasma performance, advanced plasma regimes of FFRF with a higher plasma density and fusion power of up to 100-150 MW could be possible at the later stage of FFRF without enhancement in the total plasma current.

The following subsections outline the basic properties and uniqueness of the FFRF burning plasma regime. All presented calculations have been made with ASTRA-ESC code system (IPP, Garching [13], PPPL, USA [14]). Plasma global stability margins for free-boundary magneto-hydrodynamic modes (with the toroidal wave numbers $n = 1, 2, 3$) have been calculated using KINX code [15] (CRPP, Lausanne, Switzerland, Keldysh Inst., Moscow, RF) and all presented results correspond to the plasma parameters within these margins.

Volt-second capacities of the poloidal field coil system.

The polodial field coil (PFC) system should provide activation, ramp up, and drive of the plasma current during the flat top, as well as maintain plasma equilibrium. As the reference choice for FFRF, a scaled version of the EAST PFCs was taken at the moment, as it is shown in Fig. 1(a,b).

Fig. 1(a) presents an example of initial configuration with a small plasma current (0.1 MA). The central solenoid is charged positively at maximum field of 6 T at the solenoid coils. This provides $\Psi_0 = 38.4$ Vsec of poloidal flux at the plasma magnetic axis. The reference final state of the central solenoid is chosen at magnetic field equal to -6 T. The corresponding plasma configuration is shown in Fig. 1(b). It has $\Psi_0 = -5.8$ Vsec at the plasma axis. This gives the total poloidal flux swing of 44.2 Vsec.



(a) (b)

Fig. 1. (a) Initial and (b) final magnetic configurations of FFRF.

The internal flux inside the plasma for $I_{pl} = 5$ MA is 13 Vsec. The optimal plasma current ramp up regime consumes (without non-inductive current drive assistance) about 1/3 of the internal flux, i.e. 4.3 Vsec. This leaves of $\Delta\Psi_{f-top} \simeq 40$ resistive Vsec for the flat top of the discharge.

As it will be seen in following sections, such a technically realistic flux swing can provide more than an hour of burning plasma operation and *makes FFRF independent of non-inductive current drive.*

Plasma edge and boundary conditions for core transport

Plasma edge conditions play a crucial role in energy confinement, stability and overall plasma performance. While in the conventional approach to magnetic fusion there is a strong interaction of the plasma edge with the neutral particles between the plasma and the wall, the idea of LiWF is to reduce as much as possible the cooling down effect of neutral particles on the plasma. For this purposes, the plasma particles coming out of the confinement zone are directed to the lithium covered target surface, where most of them will be absorbed. In addition, all the gas influx to the plasma edge (from the wall, gas puff, etc) should be eliminated. Instead, the NBI will provide the *core* (rather than edge) fueling of the plasma.

In the ideal situation (complete plasma particle absorption, no gas influx to the edge), the stationary plasma temperature will be uniform over the entire cross-section

$$E^{NBI} \simeq \left(\frac{3}{2}+1\right)(T_i+T_e), \qquad \frac{T_i+T_e}{2} \simeq \frac{1}{5}E^{NBI}. \quad (1)$$

This relationship is based on the fact that thermalization of the neutral beam, where the energy is contained in the ion component, is much faster than the plasma losses from the magnetic configuration. The term "3/2" in the coefficient is related simply to the definition of the temperature in terms of energy. Another term, "1", is determined by plasma losses due to diffusion and can be slightly different from 1 for a non-Maxwellian plasma.

The remarkable property of the LiWF regime is that the plasma temperature is determined exclusively by the NBI energy. Plasma physics, except for radiation, plays no role. The plasma density is determined by the beam current I^{NBI} together with plasma diffusion and is also under external control.

In the presence of an additional (to NBI) heating power P^{aux}, e.g., related to α-particles, RF, etc, or radiation P^{rad} the plasma temperature will not be entirely uniform. Still its volume averaged expression in the ideal LiWF regime will be similar

$$\left\langle \frac{T_i+T_e}{2} \right\rangle = \frac{1}{5}\left(E^{NBI}+E^{aux}-E^{rad}\right), \quad (2)$$

$$E^{aux} \equiv \frac{P^{aux}}{\Gamma^{NBI}}, \quad E^{rad} \equiv \frac{P^{rad}}{\Gamma^{NBI}}, \quad \Gamma^{NBI} = \frac{I^{NBI}}{e}, \quad (3)$$

where Γ^{NBI} is the particle source from NBI, E^{aux}, E^{rad} are introduced as specific energies of the auxiliary heating and radiation power, e is the electron charge.

In the LiWF regime, finite recycling and residual gas flux to the edge affect confinement through the boundary condition for the confinement zone. Outside the last closed magnetic surface (LCFS), the plasma energy flux $Q_{i,e}$ is delivered to the target plates convectively by the flux $\Gamma_{i,e}^{edge-wall}$ of ions and electrons

$$Q_i = \frac{5}{2}\Gamma_i^{edge-wall}T_i^{edge}, \quad Q_e = \frac{5}{2}\Gamma_e^{edge-wall}T_e^{edge}. \quad (4)$$

The subscripts 'i,e' stand for ion and electrons. Because of potentially different recycling coefficient, the ion $\Gamma_i^{edge-wall}$ and $\Gamma_e^{edge-wall}$ electron fluxes can be potentially different. In its turn, the particle fluxes to the wall consist of the diffusion flux from the plasma core $\Gamma^{core-edge} = \Gamma^{NBI}$, residual gas influx Γ^{gas} from the wall, and the flux $R_{i,e}^{ecycl}\Gamma_{i,e}^{edge-wall}$ of recycled particles from the wall, where $R_{i,e}^{ecycl}$ is the recycling coefficient,

$$\Gamma_i^{edge-wall} = \Gamma^{NBI} + \Gamma^{gas} + R_i^{ecycl}\Gamma_i^{edge-wall}, \quad (5)$$
$$\Gamma_e^{edge-wall} = \Gamma^{NBI} + \Gamma^{gas} + R_e^{ecycl}\Gamma_e^{edge-wall}. \quad (6)$$

The Eqs. (4, 6) serve as the boundary conditions for the edge plasma temperature.

The edge plasma density, on the other hand, can be determined from the following considerations. In the core, the plasma flux is determined by plasma diffusion

$$\Gamma^{NBI} = \oint_S D(\nabla n \cdot d\vec{S}), \quad (7)$$

where D is the diffusion coefficient, and integration is carried out over the LCFS. In the present studies, we neglect the favorable pinch effect. The plasma flux outside the LCFS is convective

$$\Gamma_{i,e}^{edge-wall} = \oint_S n^{edge}(\vec{V}_{i,e}^{out} \cdot d\vec{S}), \quad (8)$$

where $V_{i,e}^{out}$ are the unidirectional velocities of plasma components across the LCFS. For both electrons and ions $V_{i,e}^{out}$ can be estimated as

$$V_{i,e}^{out} \simeq V^{out} \simeq \frac{\delta_i}{\tau_i} \simeq \frac{\delta_e}{\tau_{ei}}, \quad (9)$$

where δ_i, δ_e are the characteristic banana widths of the ion and electron trajectories and τ_i, τ_{ei} is the ion-ion and electron-ion collision times. Eqs. (7, 9) with the assessment of V^{out} (9) represent the boundary condition for the edge plasma density in the LiWF regime. For a 1-D transport model it can be reduced to a form used in the simulations of this paper, i.e.,

$$n^{edge} = \frac{\Gamma^{edge-wall}D}{\Gamma^{NBI}V^{out}}n'^{edge} \simeq \frac{\Gamma^{edge-wall}}{\Gamma^{NBI}}n'^{edge}\delta_i, \quad (10)$$

where n'^{edge} is the radial gradient of the plasma density at the edge of the confinement region.

Diffusion based confinement regime

If the heat fluxes are expressed in terms of the heating power

$$Q_i + Q_i = \left(E^{NBI} + E^{aux} - E^{rad}\right)\Gamma^{NBI}, \quad (11)$$

the edge temperature will be given by

$$\frac{T_i^{edge} + T_e^{edge}}{2} = \frac{1 - R^*}{1 + \frac{\Gamma^{gas}}{\Gamma^{NBI}}}\frac{E^{NBI} + E^{aux} - E^{rad}}{5}. \quad (12)$$

where the effective recycling coefficient R^* is defined as

$$R^* \equiv \frac{R_i^{ecycl} + R_e^{ecycl}}{2} + \frac{R_e^{ecycl} - R_i^{ecycl}}{2} \times \frac{E^{aux} - E^{rad}}{E^{NBI} + E^{aux} - E^{rad}}. \quad (13)$$

For $R^* < 0.5$ and $\Gamma^{gas} < \Gamma^{NBI}$, the above written boundary conditions, in fact, constitute the really new confinement regime for magnetic fusion.

Formula (12) explicitly shows that with restriction on recycling R_i^{ecycl}, R_e^{ecycl} and the residual gas flux to the plasma edge, the high edge temperature can be indeed achieved regardless of plasma properties. Independent of plasma core properties this will automatically provide the fusion relevant plasma temperature in the entire cross-section.

The high edge temperature reduces dramatically the thermo-conduction losses of energy. It also eliminates the ion/electron-temperature gradient turbulence, which is believed to be responsible for the energy losses in the conventional plasma regimes. Concerning diffusion, the plasma ions behave neo-classically regardless of turbulence even in the present experiments.

In contrast to thermo-conduction, plasma diffusion is determined by the best confined component. Independent of the behavior of electrons, which are always anomalous, the energy losses in the LiWF regime are determined by essentially neo-classical ions. This property makes the LiWF regime the best possible regime for magnetic fusion.

As a result, the following Reference Transport Model (RTM)

$$\Gamma = D\nabla n_e = \chi_i^{neo}\nabla n, \quad (14)$$
$$q_i = n\chi_i^{neo}\nabla T_i, \quad q_e = fn\chi_i^{neo}\nabla T_e \quad (15)$$

is reasonable for LiWF regime. Here the diffusion coefficient D in the particle flux Γ is equal to the ion-neoclassical thermo-conduction value χ_i^{neo}. In the energy transport equation, the precise value of thermo-conduction coefficient is not very important for the

LiWF regime, and in the ion heat flux q_i it is equal to the same χ_i^{neo}. The electrons are assumed to be anomalous (as it is in the present experiments). In RTM, their thermo-conduction coefficient in expression for the heat flux q_e contains an extra factor f, which is scanned over the range $1 \leq f \leq 1000$.

In the following simulations, performed for FFRF, the stationary burning plasma parameters and profiles are calculated. The particle and energy source due to NBI is assumed to be parabolic.

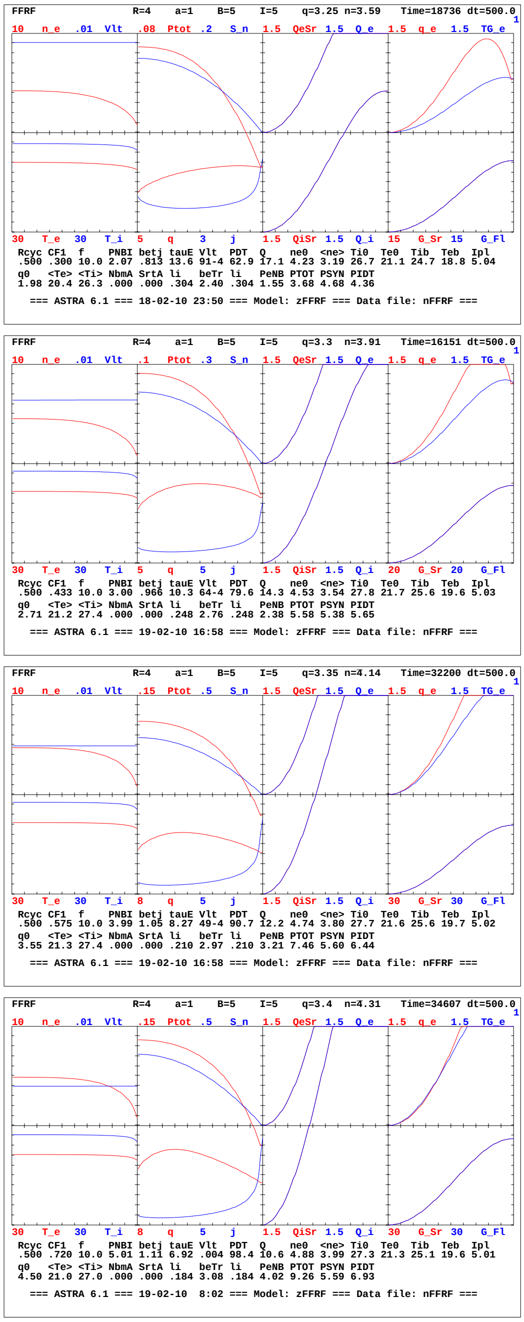
Because of the flattened current density profile in the LiWF regime, the instabilities excited by the energetic α-particles are unavoidable. In order to account for the associated energetic α-particle losses, in calculations it is assumed that only 50 % of the α-particle power is released inside the plasma. Also, because of uncertainty in energetic α-particle confinement, the calculations neglect so far plasma dilution by thermalized α-particles.

Four examples of different profiles, calculated in simulations of the burning plasma regimes are shown in Fig.2.

In each frame the abscissa is the normalized minor radius $\bar{a} \equiv \sqrt{\Phi/\Phi_0}$, where Φ is the toroidal flux through the magnetic surface. The scales of red and blue profiles are shown either at the top or at the bottom of each frame. There, $\mathtt{n_e}$, $\mathtt{Vlt}$ are plasma density and loop voltage profiles (left-top frame), $\mathtt{T_e}$, $\mathtt{T_i}$ are electron and ion temperatures (left-bottom frame), $\mathtt{P_{tot}}$, $\mathtt{S_n}$ densities of the heat and particle sources, $\mathtt{q}$, $\mathtt{j}$ are q- and $j_{\|}$-profiles. Other curves represent heat and particle fluxes in electron and ion channels. All cases are shown for recycling coefficient $R^{ecycl} = 0.5$ and electron anomaly factor $f = 10$.

The bremsstrahlung radiation is negligible in all regimes considered in this paper.

At the same time, the cyclotron radiation, as was noticed by G.Hammett (PPPL), is an important part of the burning plasma regime of FFRF. Because of good confinement, the LiWF regime does not need extra heating power in addition to NBI. At the same time, the α-particles gives 90 % of their energy to electrons, which not only do not produce fusion but can destabilize some global magneto-hydrodynamic activity. The cyclotron radiation, which in the above examples is in the range 4.5-5.7 MW, prevents overheating of electrons and keeps their temperature below the ion one. Also, the cyclotron radiation channels the α-particle energy to the side walls rather than allowing it to contribute to the heat flux into the divertor.



a) b) c) d)

Fig. 2. Output profiles from ASTRA-ESC simulations: (a) $P_{NBI} = 2$ MW, (b) $P_{NBI} = 3$ MW, (c) $P_{NBI} = 4$ MW, (d) $P_{NBI} = 5$ MW.

In present simulations the cyclotron radiation power density was calculated using a simple model, implemented in ASTRA,

$$\frac{dP_{MW}^{sync}}{dV} = 1.32 \cdot 10^{-7} (T_{e,keV} B_{tor,T})^{2.5} \cdot \sqrt{\frac{10 n_{e,20}}{a} \left(1 + \frac{18a}{R\sqrt{T_{e,keV}}}\right)}, \quad (16)$$

which reflects high sensitivity of the radiation to the electron temperature and magnetic field. Thus, the range of possible toroidal magnetic field strengths $B_{tor} = 4 - 6$ T translates into 2.75 fold change in the synchrotron radiation. The sensitivity of LiWF regime to the synchrotron radiation can be utilized purposely in FFRF for controlling the electron temperature. (Because of importance of synchrotron radiation, more accurate simulations should be performed in future using more accurate numerical models, also available for ASTRA [16]) .

As a result of synchrotron radiation and NBI fueling, in all analyzed FFRF plasma regimes, the ion temperature is higher than electron temperature. Such a "hot-ion" regime is favorable for fusion energy production as well as is consistent with the present experience with the high performance tokamak plasmas.

Fig. 3 shows the energy confinement time for different values of recycling coefficient R^{ecycl} as function of logarithm of the electron anomaly factor $f = \chi_e/\chi_i$ (abscissa axis).

The calculations in Fig. 3 demonstrate the major property of the diffusion based confinement regime of LiWF in being independent of the anomalous electron thermo-conduction.

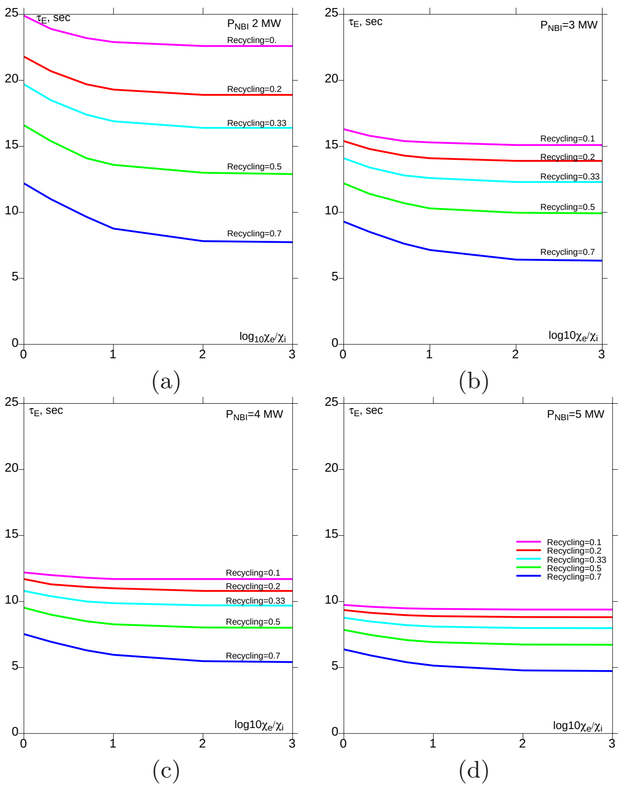


Fig. 3. Energy confinement time: (a) $P_{NBI} = 2$ MW, (b) 3 MW, (c) 4 MW, (d) 5 MW.

As soon as the recycling coefficient R^{ecycl} is noticeably smaller than 1, the energy confinement time becomes insensitive to the anomaly of electron thermo-conduction. Note, that in all other approaches to fusion, the anomalous behavior of electrons represents the major problem for plasma performance.

The green curves in Fig. 3 (and in following examples) corresponds to a realistic recycling coefficient $R^{ecycl} = 0.5$. Four cases of the NBI power, presented in Fig. 3, correspond to the same beam energy 120 kV, but different beam currents. Correspondingly, the higher NBI power, creates the higher plasma density n and reduces the energy confinement time, which in RTM is inverse proportional to n. But in all cases, the energy confinement time exceeds significantly the values typical for conventional fusion.

Fusion power

Fig. 4 shows results of calculations of fusion power P_{DT} for different values of recycling coefficient R^{ecycl} and electron anomaly factor f.

It also illustrates the dramatic drop in fusion power when the recycling coefficient exceeds 0.5 level, e.g., for $R^{ecycl} = 0.7$. This transition to low performance makes a clear distinction between the LiWF and conventional fusion regimes.

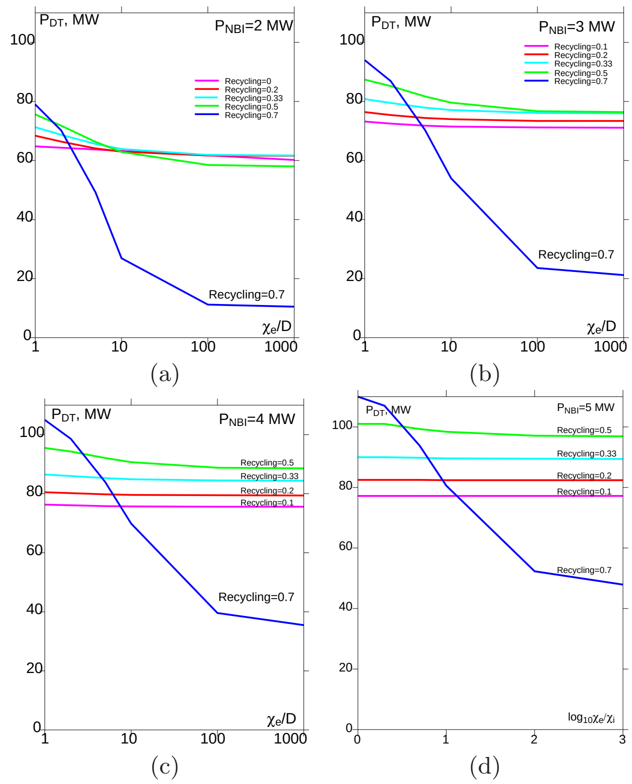


Fig. 4. Fusion power: (a) $P_{NBI} = 2$ MW, (b) 3 MW, (c) 4 MW, (d) 5 MW.

Despite reduction in the energy confinement time

τ_E at higher NBI powers, the fusion power is increasing because of enhanced plasma density, as is shown in Fig. 4. The interesting result is that the fusion power can increase with a moderate increase in recycling as in Fig. 4(c,d). This effect is explained by contribution of the enhanced edge plasma density into fusion power.

Duration of inductive current drive

Fig. 5 below gives the stationary values of loop voltage at the plasma column for different NBI powers. In all cases the loop voltage even for the $R^{ecycl} = 0.5$ case is in the range of 0.005-0.01 V, suggesting the duration of the inductive FFRF discharges in the range of 1-2 hours.

While the Volt-second capacities of the magnetic system are specified by the initial and final magnetic configurations and for FFRF is $\simeq 40$ Vsec, duration of the burning plasma regime is determined by the plasma resistivity of the flat-top of the plasma current and the bootstrap current contribution. The effect of bootstrap current should be assessed more rigorously in future.

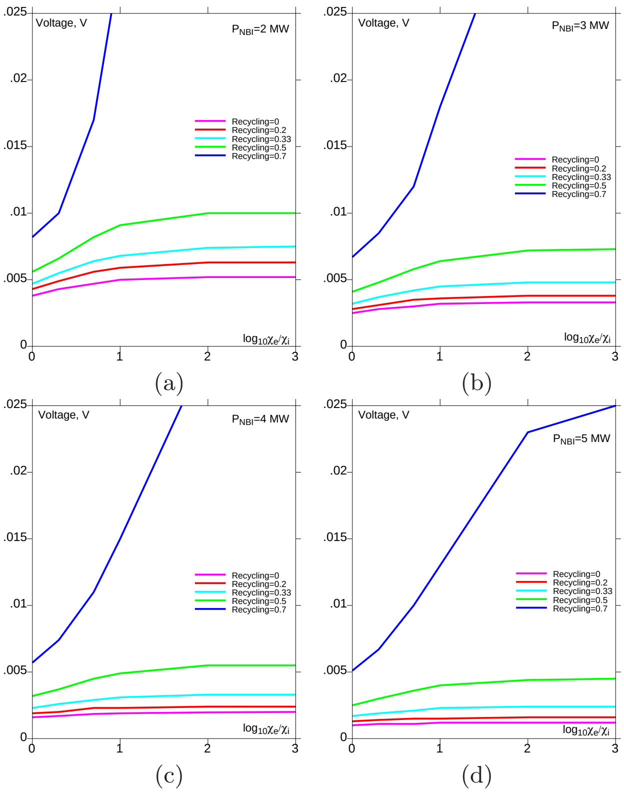


Fig. 5. Loop voltage at operational stage: (a) $P_{NBI} = 2$ MW, (b) 3 MW, (c) 4 MW, (d) 5 MW.

The possibility of a sensible pure inductive burning plasma regime, which minimizes reliance on the high-tech non-inductive current drive systems, makes FFRF especially attractive for its mission as the fusion-fission hybrid device.

Plasma stability

For the reference plasma configuration (Fig. 1(c)) the stable beta value β (the ratio of thermal and magnetic energies inside the plasma) for the global modes (with toroidal wave number $n = 1, 2, 3$) was assessed using the KINX free boundary stability code [15]. In terms of the normalized beta $\beta_N \equiv \beta_{\%} a B_{tor}/I_{pl} = 2.6$ stability margins of FFRF are not different from the present experiments. In the above given transport simulations there were no attempts to reproduce the pressure and the current profiles used in stability simulations. Nevertheless, in terms of β_N all simulations are made with $\beta_N < 2.5$ within the stability margin. The details of stability control are left for the future studies.

Concerning the plasma edge stability (Edge Localized Modes), the lithium conditioning easily stabilizes them and they do not represent a concern for the LiWF regime (unlike for the conventional approach to fusion).

Helium ash pumping

In many aspects, the LiWF regime is superior for the He ash removal from the plasma [17]. Because of core fueling and pumping edge conditions, the plasma particles, including thermalized α-particles are diffusing from the core to the edge (rather than vice-versa as in the conventional regime). Also, the LiWF regime does not need α-particle heating. In this regard, all energetic α-particle instabilities are highly beneficial for removal of α-particles from the plasma.

At the same time, much more rigorous requirements are set-up on the residual influx to the plasma edge of the He particles, which contribute to the Γ^{gas} term in Eqs. (6, 12).

The specific feature of magnetic configuration of FFRF for addressing the He ash problem is the near double null magnetic geometry with two separatrix surfaces in close proximity to each other. The inner separatrix has its open legs at the lower divertor target plates with a liquid lithium layer. The target plates absorb the heat flux, while the slowly flowing lithium absorb the deuterium and tritium from the plasma.

The helium ash is not absorbed by lithium. Instead, helium is released as low energy neutrals. Because of the magnetic mirror ratio along the field lines on the low field side of the Scrape Off Layer (SOL), there should be a blanket of trapped particles right outside the SOL. These particles can ionize the helium atoms, which will be directed along the legs of the outer separatrix to the ducts of the upper divertor with cryo-pumps. Such a scheme can potentially separate the extraction of the power and plasma particles from removal of the low energy helium ash.

The actual development of the technology for He gas pumping from the space between the plasma and the walls is a separate crucial R&D objectives for FFRF.

Plasma pumping and lithium replenishment

The NBI particle source Γ^{NBI} in FFRF is smaller than $3{\cdot}10^{20}/sec$. The residual Γ^{gas} should be reduced even to a lower level. With about 6 atomic percents of D,T solution in the liquid lithium, the requirement on lithium replenishment is only 0.05 g/sec. By itself this does not represent any challenge. At the same time, the necessary R&D should be focused on developing a stationary viscous flow of a thin lithium layer under thermal gradients, gravity, and electromagnetic force $\vec{j}\times\vec{B}$ (due to currents from the plasma to the target plates).

3 Fusion mission of FFRF

Even with reduction in requirements on plasma performance for FFH purposes, it is still necessary to make significant progress in fusion plasma R&D. The reliance of FFRF for the prevention of plasma cooling rather than on heating power is the crucial innovative element for making progress in fusion. Exceptional plasma control properties of this approach, absence of temperature gradient driven turbulence, reduced energy losses from the plasma, enhanced core and edge stability (absence of sawtooth oscillations, Edge Localized Modes and associated peaked in time thermal loads on the plasma facing components), utilization of the entire plasma volume for fusion power production, absence of the thermo-force in the Scrape Off Layer (which otherwise would drive impurities from the target plates to the plasma), consistency with non-inductive current drive methods (not necessary but potentially useful) make FFRF exceptional for a very appealing fusion mission:

1. Achieving ignition level performance in DD plasma $\langle p\rangle\,\tau_E \geq 1$ (which would be the ignition condition in the α-heated plasma) in both inductive and lower hybrid current drive regime.

2. Achieving the rate of low-density He pumping consistent with the LiWall Fusion regime.

3. Demonstrating a short (about 1min) ignition and long lasting (fraction of an hour) $Q_{DT} > 20$ in an inductively driven current regime.

4. Obtaining a long lasting (hours), or stationary, externally controlled, stable plasma regime with inductive or non-inductive (not discussed) current drive and $P_{DT} = 50 - 100$ MW.

With its fusion mission, FFRH will represent a substantial step in non-Fission Fusion (nFF) development, parallel and complementary to ITER, consistent with the on-going world fusion program.

4 Fusion-fission mission

At this time, it is not possible to specify realistically a definite mission (waste transmutation, fuel production, control of a sub-critical active fission core, etc) for a fusion-fission hybrid, which would lead either to a solution of some problems in nuclear energy, or to a better approach to them. Neither the plasma physics part of fusion, nor the blanket and tritium cycle technology are ready to offer this kind of certainty.

As a research facility, FFRF represents a necessary step for discovering the means of merging the 14 MeV fusion neutron spectrum with a variety of fission blanket compositions and regimes. In this regard, FFRF can address the following fission mission of hybrids:

1. Integrate toroidal plasma with a full size (1-1.2 m) fission blanket.

2. Develop remote handling of blanket modules situated inside the toroidal magnetic field.

3. Operate safely blankets with different content of fissile/(nuclear waste) materials at nuclear power in the range 80-4000 MW and $k_{eff} < 0.95$.

4. Operate different kinds of blankets in toroidal sectors of FFRF simultaneously.

5. Breed tritium with the use of both fusion and fission neutrons.

6. Determine practical limits on the He-cooled version of blanket.

7. Partially perform functions of a component testing facility (CTF) for the purpose of nFF development by utilizing both fusion and fission neutrons.

Utilization of a fast neutron spectrum regime in the fission blanket would be a significant enhancement in the mission of FFRF.

5 Summary

The calculations presented here demonstrate the large potential of FFRF as a neutron source for driving the fission blanket and developing the fusion-fission technology and applications. As a fusion device, FFRF is unique in its simplicity, potential performance, reliability and reliance on robust plasma physics principles and fusion technology. Although, many aspects of FFRF, including both plasma and nuclear physics still have to be analyzed in the future, the basic reference parameters are essentially determined.

The design of the tokamak core itself does not represent significant challenges and can already proceed to the conceptual design phase. On the other hand, substantial R&D is urgently necessary for Li technology, stationary NBI compatible with the neutron flux, low density helium pumping, α-particle handling technology, and for all technologies, associated with remote blanket handling inside the toroidal magnetic field. The rapid expansion of lithium conditioning research in tokamaks and stellarators, very visible at present, gives confidence in obtaining the necessary design information for FFRF in time for, at least, the fusion part of the device.

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