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Basics of Fusion-Fission Research Facility (FFRF) As A Fusion Neutron Source

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BASICS OF FUSION-FISSION RESEARCH FACILITY (FFRF) AS A FUSION NEUTRON SOURCE

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FFRF, standing for the Fusion-Fission Research Facility represents an option for the next step project of ASIPP (Hefei, China) aiming to a first fusion-fission multifunctional device [1]. FFRF strongly relies on new, Lithium Wall Fusion plasma regimes, the development of which has already started in the US and China. With R/a=4/1m/m, I_{pl}*=5 MA,* B_{tor}*=4-6 T,* P^{DT}*=50-100 MW,* $P^{fission}$*=80-4000MW, 1 m thick blanket, FFRF has a unique fusion mission of a stationary fusion neutron source. Its pioneering mission of merging fusion and fission consists in accumulation of design, experimental, and operational data for future hybrid applications.*

I. MISSION AND SCIENTIFIC STRATEGY

FFRF activity is focused on the innovative plasma regimes for the next step Chinese project of ASIPP [1] toward fusion-fission hybrids. Other activities in ASIPP, such as FDS-I,II (Fusion Driven Systems) or FDS-MFX (Multi-Functional eXperimental reactor) are concentrated on the blanket issues, while considering conventional plasma regimes as a reference case.

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 FFRF strategy is significantly different from the strategy initially adopted by the ITER project[2], which is intended to be based on ``well established data and understanding'' in plasma physics. In contrast, FFRF relies on development of plasma regimes, which emerged during the last decade (since Dec. 1998) [3-8]. The goal is to simplify the plasma regimes and eliminate numerous uncertainties in the current tokamak plasma physics.

The mission of FFRF essentially determines the major parameters of the machine. 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 plasma physics requirements limit the enlargement of the machine. The compromise solution is a major radius of about 4 m and a plasma current of about 5 MA.

These basic requirements specify the major parameters of FFRF, thus, allowing 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 a 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 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)

II. DIFFUSION BASED CONFINEMENT REGIME

For the fusion-fission device like FFRF reliable plasma control is absolutely crucial. This is the reason why conventional plasma regimes, with their numerous uncertainties in the plasma core phenomena, are not suitable for FFRF.

Fig.1 illustrates two fundamentally different confinement regimes. In the conventional case (Fig.1a), the low energy particles between the plasma core and the wall cool down the plasma edge, thus, creating a peaked temperature profile inside the plasma. In turn, the core temperature gradient leads to deterioration of energy confinement and plasma stability. It is also a source of complicated plasma physics phenomena, which are difficult to predict and control.

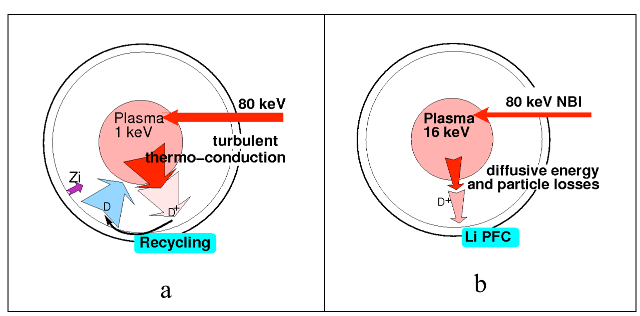


Fig 1. (a) High recycling regime of conventional fusion, and (b) the low recycling LiWall Fusion regime

The situation is much simpler (Fig.1b) if the particles from the plasma are absorbed by the wall. In this case the edge cooling is eliminated and the plasma temperature is high from the core to the edge and is determined by the energy of the Neutral Beam Injection (NBI).

For magnetically confined plasma, it is much more efficient to prevent its cooling by neutrals recycled from the walls, rather than to rely on overwhelming heating power.

A practical implementation of this understanding is now determined: a slowly moving (< 1 cm/s) thin (0.1 mm) liquid lithium layer can provide plasma particle pumping at the necessary rate. This concept, called LiWall Fusion, is adopted by the FFRF project.

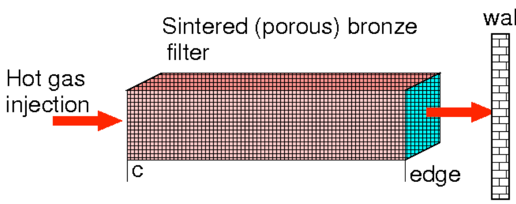


Fig.2. Sintered porous metal block illustrating different confinement regimes.

Because of the importance, we explain two confinement regimes using a more familiar example of a porous metal (Fig.2) heated by injection of a hot gas into the back surface. The sides of the block are assumed to be insulated. The analogy with the plasma confinement is pretty strong. The hot gas simulates the NBI. Porous structure mimics the magnetic field of a tokamak and eliminates free flow of the gas, thus providing confinement. Free electrons in the metal, like in a tokamak plasma, make thermal conduction large, while diffusion of the gas is much weaker and is determined by the porous size (similar to tokamaks where it is determined by the ion diffusion).

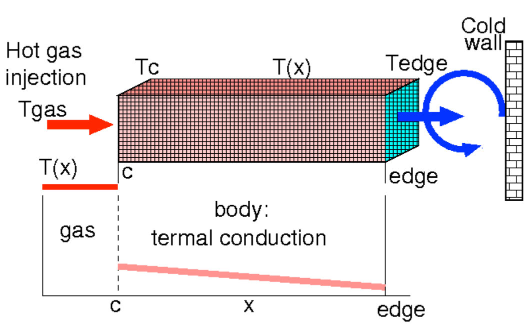


Fig.3. High recycling regime with the cold particles between the wall and the front surface.

The heating power P^{heat} is determined by the gas influx $5/2T^{gas}G^{core}$ to the back surface. The temperature T^{edge} of the front surface (replicated plasma edge) is determined by the outgoing flux $G^{edge\text{->}wall}$

$$\frac{5}{2}T^{edge}\Gamma^{edge\rightarrow wall} = P^{heat} = \frac{5}{2}T^{gas}\Gamma^{core}. \qquad (1)$$

In Fig.3 the gas particles return from the wall, and flux $G^{edge\text{->}wal,l}$ is much larger than G^{core}. As a result, T^{edge} is much smaller than the temperature of incoming gas, T^{gas}.

In contrast, in the case of a pumping wall, shown in

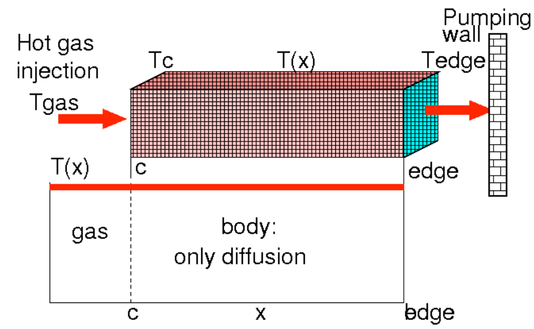


Fig.4, the temperature is everywhere equal to T^{gas}.

Fig.4. Pumping wall eliminates recycling and edge cooling.

In the first regime the confinement is low due to thermal conduction energy losses. In tokamaks, anomalous electron thermal conduction is essentially uncontrollable. In the second regime, the energy losses are determined by particle diffusion and confinement is the best possible. In the case of FFRF the confinement in the LiWall Fusion regime will be determined by the ion diffusion, which is a well confined component, behaving neo-classically.

III. PARAMETERS AND BURNING PLASMA REGIME OF FFRF

The Table 1 below specifies the reference 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	$\Delta\Psi_{f-top,Vsec}$	40
n_{20}	0.4	$\frac{T_i+T_e}{2}\|_{keV}$	24-27
$P^{NBI}_{MW}, E^{NBI}_{keV}$	120, 2-5	W_{MJ}	40
$\tau^{IND}_{E,sec}$	20-7	P^{DT}_{MW}	50-100

Table 1. FFRF parameters.

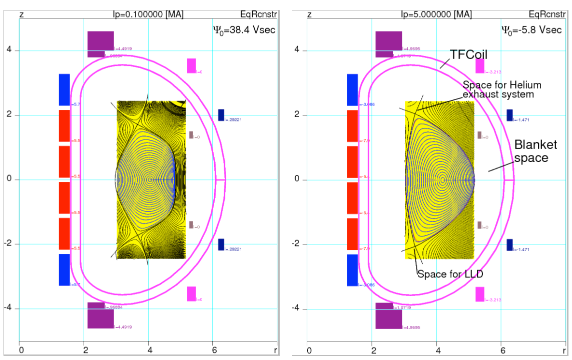
Here, a,R are minor and major radii of the plasma, V,S are its volume and surface area, n is the plasma density, E^{NBI} is the energy of Neutral Beam Injection, T_i,T_e are electron and ion temperatures, B_{tor},I_{pl} are the toroidal magnetic field (at plasma geometric center) and the plasma current, $DY_{f\text{-top}}$ the resistive Volt-second for the flat-top of the current, W_{MJ} is the total thermal energy of the plasma, t^{IND}_{E} 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 (see, e.g., [12]).

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 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.

III.A. Volt-second capacities of the poloidal field coil system.

The reference polodial field coil (PFC) choice for FFRF represents a scaled version of the EAST PFCs, as it is shown in Fig. 5.



(a) (b)

Fig.5. (a) Initial and (b) final magnetic configurations of FFRF.

Fig.5a 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 Y_0=38.4 Vsec of poloidal flux at the plasma magnetic axis. The final state of the central solenoid is chosen at magnetic field -6 T as is shown in Fig.5b. It has Y_0=-5.8 Vsec at the plasma axis. This gives the total poloidal flux swing of 44.2 Vsec. Taking into account consumption on the current ramp up, this leaves of $Dy_{f\text{-}top}$ @ 40 resistive Vsec for the flat top of the discharge.

Such a technically realistic flux swing can provide more than an hour of inductive burning plasma and *makes FFRF independent of non-inductive current drive.*

III.B. Plasma edge and boundary conditions for core transport

As was shown earlier, plasma edge conditions play a crucial role in energy confinement, stability and overall plasma performance. For the tokamak plasma Eq.(1) has

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

its analog for the edge ion and electron temperatures Here, G^{NBI} is the particle source from NBI, $E^{aux}=P^{aux}/G^{NBI}$, $E^{rad}=P^{rad}/G^{NBI}$, P^{aux}, P^{rad} are auxiliary heating and radiation powers, G^{gas}, is the residual particle flux, other than recycling, to the plasma edge. The recycling coefficient R^* is defined in terms of partial recycling coefficients R_i, R_e of ions and electrons as

$$R^*\equiv\frac{R_i+R_e}{2}+\frac{R_e-R_i}{2}\frac{E^{aux}-E^{rad}}{E^{NBI}+E^{aux}-E^{rad}}. \quad (3)$$

In the case of heating/fueling by NBI and in the absence of recycling and gas influx the plasma edge temperature is simply $E^{INB}/5$. For E^{INB}=120 keV it is 24 keV.

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. Besides diffusion, the plasma density is determined by the beam current I^{NBI} and is under external control.

In its turn, the edge plasma density is determined by a unidirectional particle flux from the edge

$$n^{edge}=\frac{1+\frac{\Gamma^{gas}}{\Gamma^{NBI}}}{1-R^*}n'^{edge}\delta_i, \quad (4)$$

where d_i is a characteristic diffusion step size of the order of the poloidal larmor radius or banana width.

The boundary conditions (2-4), describing the LiW Fusion regime in FFRF, represent a new element in the tokamak transport simulations.

III.C. Energy confinement and Fusion power

The following Reference Transport Model (RTM) is appropriate for the diffusion based confinement regime

$$\Gamma=D\nabla n_e=\chi_i^{neo}\nabla n, \quad (5)$$

$$q_i=n\chi_i^{neo}\nabla T_i, \quad q_e=f^{anom}n\chi_i^{neo}\nabla T_e \quad (6)$$

Here the diffusion coefficient D in the particle flux G is equal to the ion-neoclassical thermal conduction value $c^{neo}_{i.}$ In the energy transport equation, the precise value of thermal conduction coefficient is not very important for the LiWF regime, and in the ion heat flux q_i it is equal to the same $c^{neo}_{i.}$. The electrons are assumed to be anomalous (as it is in present experiments). This is reflected by a factor f^{anom} scanned in the range $1 < f^{anom} < 1000$.

Fig. 6. shows an example of ASTRA-ESC simulations of the burning plasma regime in FFRF. In order to account for the associated energetic a-particle losses, in calculations it is assumed that only 50 % of the a-particle power is released inside the plasma. In each frame the abscissa is the normalized minor radius

a=sqrt(F/F0), where F is the toroidal flux through the magnetic surfaces. The scales of red and blue profiles are shown either at the top or at the bottom of each frame. The recycling coefficient R*=0.5 and electron anomaly factor f^{anom}=100.

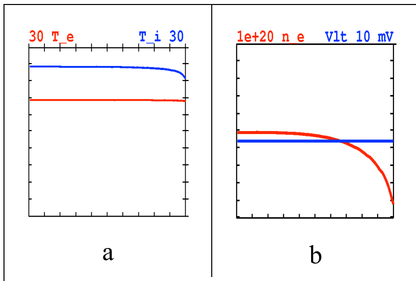


TABLE 2.

Parameter	Value
P^{DT}_{MW}	97.1
P^{NBI}_{MW}	5
$P^{DT}_{i,MW}$	6.73
$P^{Cycl}_{i,MW}$	5.39
$\tau^{IND}_{E,sec}$	6.7
$T_{i,keV}$	26.6
$T_{e,keV}$	20.7
n_{20}	0.4

Fig.6. (a) Ion (blue) and electron (red) temperature profiles, (b) plasma density (red) and loop voltage (blue) profile.

The loop voltage level of 5 mV (Fig6.b) suggests duration of the burning plasma with an inductive current drive of more than 2 hours.

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.

The bremsstrahlung radiation is negligible in FFRF regimes. At the same time, the cyclotron radiation is an important part of the burning plasma regime of FFRF. In present simulations the cyclotron radiation power density was calculated using a simple model

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

The cyclotron radiation, which in the above example is 5.4 MW, prevents overheating of electrons and keeps their temperature below the ion one.

The "hot -ion" burning plasma regime is a unique property of the LiW Fusion regime of FFFR, favorable for both fusion production and plasma stability.

Fig.7 shows the energy confinement time and fusion power for different values of recycling coefficient R^{ecycl} as function of the logarithm of the electron anomaly factor $\log_{10} f^{anom} = \log_{10} c_e/c_i$.

As soon as the recycling coefficient R^{ecycl} is less than 0.5, the energy confinement time and fusion power is insensitive to the anomaly of electron thermal conduction.

At the same time a dramatic drop in fusion power with increased f^{anom} is visible when the recycling coefficient R=0.7. This transition to low performance makes a clear distinction between the LiWF and conventional fusion regimes.

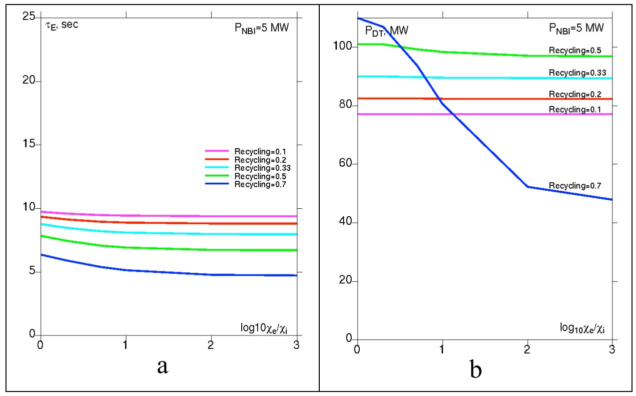


Fig.7. (a) Energy confinement time and (b) fusion power for P_{NBI}=5 MW, E_{NBI}=120 keV.

III.D. Plasma stability

For the reference plasma configuration (Fig.5b) the stable beta value b (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 b_N=b% aB_{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 b_N all simulations are made with $b_N < 2.5$ within the stability margin. The details of stability control are left for future studies.

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

III.E. Helium ash pumping

In many aspects, the LiWF regime is superior for He ash removal from the plasma [16]. Because of core fueling and pumping edge conditions, the plasma particles, including thermalized a-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 a-particle heating. In this regard, all energetic a-particle instabilities are highly beneficial for removal of a-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 G^{gas} term in Eqs. (2,4).

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 objective for FFRF.

III.F. Plasma pumping and lithium replenishment

The NBI particle source G^{NBI} in FFRF is smaller than 3e+20/sec. The residual G^{gas} should be reduced to an even lower level. With about 6 atomic percents of D,T solution in the liquid lithium, the requirement on lithium replenishment is only 0.05g/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 $\mathbf{j}\times\mathbf{B}$ (due to currents from the plasma to the target plates).

IV. 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 $\langle p\rangle t_E=1$ (which would be the ignition condition in the a-heated plasma) in both inductive and lower hybrid current drive regimes.
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 be complementary to ITER, consistent with the on-going world fusion program.

V 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.

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.

VI. 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, a-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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