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COMPACT TORI FOR ALTERNATE-FUEL FUSION

George H. Miley
Fusion Studies Laboratory
University of Illinois
103 S. Goodwin Avenue
Urbana, IL 61801 USA

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George H. Miley
Fusion Studies Laboratory
University of Illinois
103 S. Goodwin Avenue
Urbana, IL 61801 USA

ABSTRACT

A D-T burn dynamic experiment using a scaled-up field-reversed theta-pinch device is proposed as a first step towards demonstration of a D- ^3He burn. The feasibility of such an experiment is based on a loss-cone-like confinement scaling described here plus the concept of using combined pellet and neutral-beam injection to heat the plasma build-up, its density, and damp the $m=2$ instability. The potential extension to a D- ^3He reactor is also discussed.

1. Introduction

Advanced-fuel fusion offers many potential advantages such as: the elimination of the need for tritium breeding; reduced neutron fluxes, i.e., reduced materials damage; and improved energy conversion using the larger charged-particle yield (see Ref. 1 and references therein). However, fuels such as p- ^{11}B that offer the most benefits do not appear usable with presently known confinement approaches.

Still many advantages might be obtained using nearer-term deuterium-based fuels, especially: catalyzed-deuterium (Cat-D), semicatalyzed-deuterium (SCD), and D- ^3He , all of which are independent of tritium breeding. A particularly interesting scenario would be extract ^3He from the exhaust plasma from a SCD plant and ship the ^3He to satellite D- ^3He units⁽²⁾. The SCD plant could produce electricity and

it also appears well suited for portable ("synthetic") fuel production, e.g. via high-temperature electrolysis of water to produce H_2 (3). The $D-^3He$ satellites could benefit from their good environmental characteristics to achieve plant siting near the user, thus serving as "distributed" electrical producers. This combination would yield about 70% of the energy produced as portable fuels and 30% as electrical energy, a fairly ideal mix according to projected needs for the 21st century.

These thoughts seem quite idealistic today as we are still struggling to demonstrate scientific feasibility with D-T fuel. However, once a D-T burn is achieved, if it is used in a device with favorable confinement scaling, thermal run-away to the temperatures required for SCD or $D-^3He$ operation (~40-50 keV) may be straightforward. In fact, control systems appear necessary to prevent run-away if optimum D-T operation is desired.

To be of eventual use as an advanced-fuel reactor, the confinement scheme selected should have a high β in order to reduce radiation losses and achieve competitive power densities. Both features--increasing $n\tau$ with temperature and high β --are in varying degrees, characteristic of compact tori concepts such as the Spheromak, Field Reversed Mirror (FRM) and Field Reversed Theta Pinch (F RTP).(4) Thus, an essential element in the development of advanced fuels would appear to be the timely development of an experimental compact tori facility capable of burn dynamic studies, starting with D-T and eventually extending to $D-^3He$ burns. In the present discussion we consider the possibility of using a F RTP for such a facility for several reasons: 1) present F RTP experiments using the FRX device at the Los Alamos National Laboratory are encouraging(5), 2) the loss-cone scaling relations recently proposed by Fang and Miley(6-8) compare well with experiments and predict favorable scale-up for a burn-dynamic-type experiment 3) as proposed here, the use of neutral-beam and pellet

injection, employing state-of-the-art technology, would allow rapid scale-up from present experiments, and 4) preliminary conceptual studies indicate that the extension to advanced fuel-reactors is potentially attractive.

2. The Field-Reversed Theta Pinch (F RTP)

The F RTP concept is illustrated schematically in Fig. 1. The closed magnetic field geometry is formed within an open-field region using a programmed pinch implosion where the reversed fields initially created are followed by reconnection of the ends of the inner field due to tearing induced by stronger pinch coils at the end of the main coil(5,9,10). For a reasonable range of total energy (H) and canonical angular momentum (P_θ), ions within the closed-field region are absolutely confined by the magnetic potential well. However, due to the relatively large orbits involved, ions with certain values of H and P_θ are not confined and may escape along open field lines(11,12). These considerations then define a loss region in (H, P_θ) space, and an important loss mechanism occurs by scattering into this loss region, (called Loss-Cone Scattering Transport, LCST). Indeed, our studies(6-8) indicate that this loss rate dominates over classical pressure-gradient-driven cross-field diffusion. There is some evidence though that in present experiments lower hybrid drift (LHD) wave instabilities due to the steep density gradients near the separatrix result in an even larger loss rate(5,13). However, it may be possible to control these instabilities by tailoring the density profile, and indeed this appears necessary for favorable scaling to the reactor regime. Consequently, in the present study, we assume that LCST represents the proper scaling relation for use in the design of an ignition experiment of the type envisioned here.

3. LCST Theory

Available experimental data is too limited to predict confinement scaling for the F RTP over the range desired in the present study. Nor is

Figure 1. Geometry of F RTP configuration.

there any basis to justify use of scaling laws for other devices, e.g. the widely used Alcator scaling derived from tokamak experiments. Consequently, a fundamental basis for this study is the formulation of a LCST scaling law. The basic LCST theory is briefly outlined below.

The physical basis for LCST theory is that in the reversed-field geometry, particles having a prescribed range of energy and canonical angular momentum will be absolutely confined, but particles falling out of this range can escape from the system along open-field lines (see Fig. 2). Then LCST theory assumes that particle losses are dominated by collisions that result in confined particles changing their energy-momentum such that they escape from the closed-field region. In other words, scattering results in a diffusion in (H, P_θ) space across the confinement boundary, i.e., the process is similar to scattering into the loss cone in a simple mirror. However now, as shown in Ref. 12, for certain limits the confined region is a sphere in velocity coordinates. Hence the term "loss-cone-like" has been adopted.

By definition the particle confinement time for species α , (α = ions, electrons), designated by $\tau_{p\alpha}$, is

$$\tau_{p\alpha} = - N_\alpha / (dN_\alpha / dt) \quad (1)$$

where N_α is the total number of particles present in the plasma volume V and the loss rate dN / dt is given by

$$\frac{dN_\alpha}{dt} = - \int dV \sum_\beta \oint_{S_v} \underline{j}^{\alpha/\beta} \cdot d\underline{S} \quad (2)$$

Here S_v is the boundary surface for losses in velocity space, $d\underline{S}$ is a surface element on this boundary, and $\underline{j}^{\alpha/\beta}$ is the particle current in velocity space due to collisions of species α with species β (where β includes α).

Figure 2. Plausible particle paths leading to axial losses. Paths B, C and E are most probable, while Paths A, D and F rarely occur due to the higher energy required.

In the present case, the approximate loss boundary in (H, P_θ) space ⁽¹²⁾ is transformed to (v_r, v_θ, v_z) and can be shown ^(8,11) to be a sphere of radius v^* centered at $v_\theta = -r \omega_0$ where:

$$v^* = ((-2q_\alpha(\omega_0\psi + \phi) + m_\alpha r^2 \omega_0^2)/m_\alpha)^{1/2}, \quad (3)$$

Here

$$\omega_0 \equiv q_\alpha B_0/m_\alpha \quad (4)$$

where B_0 is the magnetic field in the end throat region of the pinch (Fig. 3) and q_α and m_α are the charges and mass, respectively, of the α species particle. Thus ω_0 is the gyrofrequency at the end throat while ψ and ϕ are the magnetic flux function and electric potential at position r respectively. Note that the loss boundary depends on radial position r through Eq. (3).

The current $j^{\alpha/\beta}$ required in Eq. (2) is given by:

$$j_\ell^{\alpha/\beta} = \frac{F_\ell^{\alpha/\beta}}{m_\alpha} f_\alpha - \sum_k D_{\ell k}^{\alpha/\beta} \frac{\partial f_\alpha}{\partial v_k} \quad (5)$$

where ℓ indicates the velocity coordinate, $F^{\alpha/\beta}$ and $D^{\alpha/\beta}$ are the dynamic friction and diffusion tensors, respectively, and f_α is the distribution function for species α . The LCST approximation assumes that the velocity distribution function can be represented as an exponential rigid rotor modified to vanish on the boundary S_v given by Eq. (3), i.e.,

$$f(v) \propto [\exp(-m(v^2 - 2v_0 v \sin\theta \sin\phi + v_0^2)/2T)](1 - v^2/v^{*2}) \quad (6)$$

$$v_0 = r(\omega_0 + \omega) \quad (7a)$$

$$v = (v_r^2 + v_z^2 + v_\theta^2)^{1/2} \quad (7b)$$

$$v_\theta = v \sin\theta \sin\phi \quad (7c)$$

Figure 3. Magnetic field structure in the r - z plane. In a future D-T pilot unit an appropriate first-wall, blanket, and shield plus divertor "energy dumps" would be added.

where we have adopted spherical velocity coordinates (v, θ_v, ϕ_v) centered at $-r\omega_0$, ω being the rotation frequency. The confinement region defined by S_v is shown in Fig. 4 along with the corresponding region in H, P_θ space. These relations, i.e., Eqs. (1-7), are used to calculate the confinement time $\tau_{p\alpha}$ using the equilibrium density profile $n(r)$, magnetic flux function $\psi(r)$, and electric potential profiles $\phi(r)$ consistent with the modified rigid rotor assumption for f , namely

$$n(r) = n_m \operatorname{sech}^2 \xi, \quad (8a)$$

where

$$\xi = K(r^2/R^2 - 1), \quad (8b)$$

$$n_m = n_0 / \operatorname{sech}^2 K. \quad (8c)$$

Here n_0 is the particle density at the system axis ($r=0$) while the maximum density n_m occurs at the plasma major radius R . The "reversal" factor K is given by:

$$K = -\tanh^{-1}(B_{(r=0)}/B_{\text{ext}}), \quad (9)$$

while the flux function $\psi(r)$ is given by

$$\psi(r) = \frac{T_i + T_e}{e(\omega_e - \omega_i)} \left[2 \ln \left(\frac{\cosh K(r^2/R^2 - 1)}{\cosh K} \right) - r^2 \frac{m_i \omega_i^2 + m_e \omega_e^2}{2(T_i + T_e)} \right]. \quad (10)$$

Figure 4. Confinement regions: a) in H - P_θ space, and b) in velocity space for a specific radial position $0 < r < r_w$. Losses occur by scattering and transport across the spherical surface in (b) which, in (a) corresponds to leakage across the straight line representing the absolute confinement boundary.

The radial electric field E_r , hence , is given by

$$E_r = - \frac{\partial \phi}{\partial r} = \frac{4K T_i T_e}{e R^2 (\omega_i - \omega_e)} \left(\frac{\omega_i}{T_i} + \frac{\omega_e}{T_e} \right) r \tanh \xi$$

$$+ r \frac{\omega_i \omega_e (m_i \omega_i + m_e \omega_e)}{e (\omega_i - \omega_e)}$$
(11)

where it is assumed that the total field is of the form $\vec{E} = E_r \hat{r}$. In solving these relations for τ_p , the ion temperature T_i , electron temperature T_e , major radius R , reversal factor K and external magnetic field B_{ext} are treated as input parameters. The plasma length L , selected to satisfy axial pressure balance, enters in the determination of the total number of particles in the plasma.

A test of this theory is shown in Table I where confinement times predicted by LCST theory are compared with reported τ_p 's (or, alternately, the "stable" time τ_s , which is in general of order of τ_p). Reasonable agreement is seen for cases other than for the experiment by Eskov, Kurtmullaev, et al.(10) which appears to have been limited by the magnetic field decay time rather than particle losses. (As discussed in the following section, the relatively large radius of this device apparently resulted in an exceptionally large τ_p .)

4. Scaling Predictions

The numerical results from τ_p calculations have been correlated as a function of five key parameters: the ion temperature T_i , external field B_{ext} , peak density n_m , field coil size r_c , and reversal factor K . The effect of electron temperature was neglected as a weak dependence and R was eliminated in favor of r_c using the following expression based on radial and axial pressure balance :

$$\frac{R^2}{r_c^2} = 1 - \frac{\tanh K}{K} .$$
(12)

Table I Operating Parameters and Predicted Confinement Times

Parameters	Kolb, et al. Ref. 14	Eberhagen & Grossman Ref. 15	A.G. Es'kov, et al. Ref. 10	FRX-A Refs. 5,9	FRX-B Refs. 5,9	TRX-1 Ref. 16
External Field, B_{ext} (T)	6.3	1.4	1.0	0.68	0.75	0.55
Peak density, n_m (10^{15}cm^3)	50	44	8.3	3.65	4.1	2.5
Ion temp., T_i (eV)	1000	60	100	185	240	200
Electron temp., T_e (eV)	1000	50*	200	120	100	100
Major radius, R (cm)	0.7	1.4	11.0	4.0	4.0	4.95
Plasma length, L (cm)	180	40	150	35	36	80
Field coil radius, r_c (cm)	7*	7*	23*	12.5	12.5	12.0
Reversal factor, K	0.6	0.8	0.97	0.6	0.6	0.8
Line density, N_L (10^{17}cm^{-1})	1.4	4.5	48.0	2.7	1.9	3.2
Decay time of B_{ext} , τ_B (μs)	100	100	200	100	150	**
Obs. stable time, τ_s (μs)	15	7-13	100	15-20	20-40	40-50
Measured τ_p (μs)	**	**	**	30-50	40-80	~ 50
LCST Prediction, τ_p (μs)	13	5	450	41	53	78

* Estimated

** Not reported

The scaling relation obtained is

$$\tau_p \sim T_i^{0.2} K^{2.7} r_c^{1.8} B_{ext}^{2.5} n_m^{-1} \quad (13)$$

As seen from Table I, this result is consistent with experiments having a wide range of parameters. Recent experiments on the FRX-A and B devices (with r_c = fixed) suggest that τ_p is proportional to R^2/ρ_i (where ρ_i is the ion gyroradius) and that τ_p has a very weak temperature dependence. The strong dependence on the plasma radius is perhaps the easiest parameter at the disposal of someone designing an experiment to obtain improved confinement. For r_c constant, R is approximately proportional to K , consequently, Eq. (13) suggests $\tau_p \sim R^{2.7}$ which is a somewhat stronger dependence than reported by the FRX team. This difference could well be due to various simplifying assumptions made in the LCST theory, or due to uncontrolled factors in the experiment. Still, based on Table I, Eq. (13) is thought to provide a reasonable first estimate of scaling for the FRTP. In addition, this same LCST model predicts the plasma rotation observed in FRTP experiments reasonably well.⁽¹⁷⁾ In this model spin-up is associated with preferential angular momentum carried by escaping particles while the leakage rate is described by the LCST theory outlined above.

5. Burn Dynamic Pilot Unit

The first step towards a D-³He experiment would be the development of a pilot unit capable of studies of D-T burn dynamics, i.e., the ability to use D-T thermal runaway to achieve the higher temperatures necessary for D-³He. This pilot unit would, as suggested earlier, employ a combination of neutral-beam and pellet injection to heat and build up the target FRTP plasma density.

A strategy for designing the pilot unit is illustrated on the $n\tau_E$ vs T_i diagram of Fig. 5. Here the various straight lines represent "operating" $n\tau_E$ curves based on the scaling of Eq. (13) for a device with a 24-kG external field, a unit reversal factor ($K=1$) and various magnet coil radii ranging from 0.4 to 0.8 meters. (The choice of $K=1$ allows some conservatism in the design since, with good operation, larger values of K should be obtained) The curved line represents the "plasma $n\tau_E$ corresponding to a break-even energy balance, i.e.:

$$(n\tau_E)_{PL} = 3kT/(1/4 \langle \sigma v \rangle E_\alpha - P_R) \quad (14)$$

where E_α is the α particle energy and P_R is the total power loss by radiation, including line radiation, Bremsstrahlung, and cyclotron emission. For the curves shown, we have assumed $T_e=T_i$ and used a cyclotron radiation coefficient⁽¹⁸⁾ $K_C = 5 \times 10^{-3}$, corresponding to a graphite-coated metallic first wall. Based on results from the FRX experiments^(5,9) we assume that $\tau_E \sim .8 \tau_p$. The LCST theory predicts that the energy spectrum of escaping particles are very close to that of the confined distribution. Consequently, this approximation is thought to be reasonably accurate.

A direct result of LCST is that $(n\tau_E)_{OP}$ is only weakly dependent on temperature. (For comparison, note that corresponding $(n\tau_E)_{OP}$ curves for a simple mirror would scale as $T_i^{3/2}$ while curves for a tokamak would scale approximately as $T_i^{1/2}$ -- see Ref. 18). Consequently, a higher $n\tau$ is most easily achieved by selecting a larger plasma radius, i.e., a larger magnet coil radius r_c . For the illustration in Fig. 5, a coil radius of 0.6m is selected as the operating point. This provides breakeven operation at $T_i \sim 15$ keV and also allows studies of thermal run-away up to the higher temperature intersection at $T_i \sim 35$ keV, i.e., in the region of interest for burning D-³He. In the actual design presented later, a slightly smaller radius, unequal ion-electron

Figure 5. $(n\tau_E)_{OP}$ and $(n\tau_E)_{PL}$ versus T_i for a D-T system. Here $(n\tau_E)_{OP}$ is based on LCST scaling while $(n\tau)_{PL}$ corresponds to energy breakeven.

temperatures, and other minor changes are indicated, based on a more detailed calculation.

A key aspect of the present concept is the use of neutral-beam and pellet injection. In addition to the usual functions of heating and refueling, with proper orientation, the momentum associated with the injected neutral beam can help suppress the $m=2$ rotational instability.⁽⁷⁾ Likewise, pellet injection can be designed to maintain the plasma pressure profile near its equilibrium and, in the process, provide diamagnetic currents that aid in maintenance of field reversal⁽¹⁹⁾. Other studies of possible approaches to RFTP scale-up have generally avoided use of neutral beams in favor of compression heating. However, this requires translation into separate chambers and good control to avoid instabilities during periods of small plasma/wall radius ratio. The neutral-beam approach selected here in many respects represents a "brute force approach" to achieve definitive heating while maintaining maximum control. Indeed the combination of neutral beams plus pellets provides a flexibility that seems desirable when moving into new regimes represented by the measured experiment.

The design parameters for the proposed F RTP device are summarized in Table II and are compared with other devices, namely TRACT⁽²⁰⁾ (a small F RTP similar to the present concept in many ways but without neutral beam or pellet injection), a beam-driven field-reversed mirror (FRM)⁽²¹⁾, and a FED size tokamak⁽²²⁾. The present device would use a 24-KG external field, a 0.29-m plasma with density $\sim 3 \times 10^{20} \text{m}^{-3}$ at 15 keV. Neutral-beam injection plus the start-up implosion energy would provide the equivalent of 36-MW input power (P_{in}) and result in a fusion

Table II Comparison of the Proposed F RTP with TRACT and
Other Burn Experiments

Parameters	F RTP Concepts		Other Concepts	
	Present	TRACT Ref. 16	FRM Ref. 19	Tokamak Ref. 22
R (m)	0.29	0.24	1.06	4.8
a (m)	0.11	-	-	1.3
L (m)	10.0	2.1	-	-
r_w (m)	0.45	0.5	(1.65)	1.5
T_i (keV)	15.0	15.0	17.5	10.0
B_{ext} (kG)	24.0	70.0	15.0	36.0*
$\bar{n}$ ($10^{21} m^{-3}$)	0.3	2.0	0.1	0.08
P_{in} (MW)	36	†	10	36
P_F (MW)	106	(550)	104	180
P_{NW} (MW/m ²)	2.0	15.0	2.0	0.4
$\langle \beta \rangle$	0.76	(.5)	<0.5	0.055
τ_{burn} (s)	4.5	0.5	**	100
P_D (MW/m ³)	15.0	(260)	5.5	0.7
τ_E scaling	$0.8\tau_p$ (LCST)	$1/30\tau_{class}$	Alcator	Alcator
Q	2.4	**	(10)	5
P (MW)	26	90	**	72

* field on axis

** data not given

† 80-MJ magnetic compression energy

() estimated values

power of ~ 100 MW over the burn time of 4.5 sec. A major difference compared with TRACT is that the present design has a neutron wall-loading (P_{NW}) of ~ 2 MW/m² vs the very high 15 MW/m² of TRACT. This avoids some of the maintenance problems encountered in TRACT. While there are also similarities to the FRM device, the current experimental data base for the F RTP is preferable should such a scale-up be desired. As expected, the F RTP is considerably smaller in physical size than is the FED-type tokamak, although the net power outputs only differ by approximately a factor of three. Thus the F RTP has several obvious advantages--reduced capital costs and faster scale-up (fewer intermediate size devices required) -- assuming that the various physics assumptions and scaling used here are valid.

Several points about the F RTP concept need elaboration. First, for simplicity, the burn is assumed to take place within the implosion chamber. However, if it turns out to be necessary, the target plasma could be translated (using programmed magnetic fields) into a second chamber prior to injection heating and burn. Two issues are involved in the feasibility of having an in-situ burn in a single chamber: maintenance of the first wall and space for injection ports. The first wall issue is not considered to be serious since as pointed out earlier the wall loading has purposely been held down in this design. Further, in an experiment, long lifetimes are not as important as in a power reactor. The second issue, space for penetration of injection ports, also appears manageable. At the densities involved, a fairly high neutral-beam energy (~ 100 keV) can be used while keeping a good trapping efficiency ($>85\%$).

With the 100-keV neutral-beam injection-energy selected (cf. 15 MW of heating power), a beam current of 150 A is indicated. If we assume a current density limit of 0.2 A/cm^2 for the neutral beam, a cross section of 730 cm^2 is needed, which is about 15% of the plasma end surface area (tangential injection near the plasma ends is envisioned to control spin-up). Despite the complex implosion coil structure, space of this magnitude appears to be available.

The neutral-beam current of 150 A corresponds to a refueling rate of $\sim 0.9 \times 10^{21} \text{ ion/sec}$. This is only about 12% of the total refueling rate required, the remainder being supplied by pellet injection. Still the angular momentum carried by the neutral beam is about 40-60% of the background ions, providing a significant control over excessive spin-up. Indeed, a modest rotation velocity is thought to be desirable in order to stabilize the plasma against the tilting mode instability.(23,24) However, as occurs in present experiments, too high a rotation velocity can cause plasma breakup.

The time sequence of the initial heating and burn is illustrated in Fig. 6. The implosion process is assumed to bring the plasma temperature to T_{im} which is $\sim 20\%$ of the final desired value, T_f . The overall efficiency for the implosion process, defined as the ratio of the plasma thermal energy to the total stored capacitor bank energy, is assumed to be $\sim 1\%$ based on FRX experience. The burn time, τ_{burn} , is selected to be $\sim 5 \tau_p$. This is adequate to provide data on both thermal run-away dynamics and other important processes such as ash build-up, plasma-wall interaction, etc. The open-field line region (Figs. 1-3) provides a natural divertor for exhaust plasma, both during operation and for the quench at the end of the burn. Thus, a plasma energy "dump" is planned in place of the end walls shown in Fig. 3. Since this dump must handle a large power loading, a special design such as the lithium "fall" described previously(20) would be necessary and would require a separate development program.

Fig. 6. An assumed ion temperature history for the FRTP burn experiment.

6. Burn Dynamic Experiments

The pilot unit described in the preceding section would mainly be used to establish $n\tau_E$ scaling at higher temperatures. Thus, as indicated in connection with Fig. 5, the design is intended to allow excursions up to temperatures of order of 30-40 keV. This would be done with a programmed decrease in density (largely controlled by pellet injection) in order to maintain pressure balance without excessive plasma expansion. This would also prevent the increased radiation loading (esp. cyclotron radiation) from exceeding thermal limits ($\sim 1.0 \text{ MW/m}^2$) on the first-wall surface. A distinct advantage of the proposed pilot unit is that the combination of neutral beams, pellet injectors, and natural divertor-high power collection provide the degree of flexibility and control necessary for dynamic experimentation.

Once high temperature experience is obtained with D-T excursions, a decision on how to extend the burn to D- ^3He (i.e., replace tritium as it fuses with ^3He) would be necessary. If $n\tau_E$ scaling turned out to be sufficiently favorable to reach D- ^3He requirements in the pilot unit, the problem would resolve into the development of appropriate fueling techniques to add ^3He during the burn.

$(n\tau_E)_{PL}$ requirements for D- ^3He are about an order of magnitude higher than for D-T shown in Fig. 5. However, successful operation of the pilot unit with larger reversal factors, e.g., $K \sim 2.5$ vs the $K=1$ assumed in Fig. 5, could possibly make up this deficit, (cf Eq. (13)). A low rate of ^3He fueling could be obtained by designing the neutral-beam injectors to handle both tritium and ^3He (see Ref. 25 for design considerations related to a ^3He neutral-beam injector). Pellet fueling with ^3He would be necessary for higher fueling rates, but this poses the difficult challenge

of developing appropriate ^3He pellets, e.g., a deuterium-shell type noted in Ref. 26. Gas puffing could provide an alternative approach. Then the maintenance of the desired "quasi-steady" state pressure profile would have to rely solely on pure deuterium pellet injection, the ^3He influx coming from gas puffing.

If the $n\tau_E$ scaling turned out to be less favorable, not only would it be necessary to develop ^3He refueling techniques but a second generation pilot device would be required. Following the scaling of Eq. (13), this would probably involve the combination of a larger plasma (hence coil) radius and stronger external fields. This would be more expensive and require a larger capacitor bank, but in view of the rather modest parameters of the pilot unit (cf Table II), such a scale-up should be feasible.

7. Reactor Extension

As stressed earlier, an important added motivation for considering a F RTP ignition device is that, in addition to the data provided by the experiment itself, this approach also potentially provides a route to attractive advanced-fuel power reactors. By virtue of the high β offered by these reactors, fueling with either Cat-D or D- ^3He appears feasible. To verify this, we have carried out scoping studies of several conceptual prototype reactors based on the LCST scaling, neutral-beam injection and "bootstrap" heating (i.e., D-T startup for Cat-D and D- ^3He devices). The general ground rules for these studies are listed in Table III.

Table III PROTOTYPE FRTP REACTOR ASSUMPTIONS .

- STORED ENERGY FOR IMPLOSION $\bar{<}$ 85 MJ
- IMPLOSION EFFICIENCY, 2%
- NEUTRAL-BEAM INJECTION POWER, $<$ 50 MW
- COIL-FIRST WALL SEPARATION, 0.15-0.75m
- EXTERNAL FIELD, $<$ 10 kG
- NEUTRON WALL LOADING, $<$ 2 MW/m²
- RADIATION WALL LOADING, $<$ 1.5 MW/m²

Results from these scoping studies are shown in Table IV. Operational temperatures for Cat-D and D-³He are set at 40 and 50 keV respectively. Both employ a coil/wall radius ratio of 0.6m/0.45m and a major radius/length product of 0.3m x 4m, giving a plasma volume of 2.1 m³. The magnetic field is increased to the maximum value allowed by the wall loading in each case (86 and 100 kG for Cat-D and D-³He respectively). Note that the radiation wall loading represents the limiting feature in the D-³He case while the neutron wall loading dominates for D-T and Cat-D. Indeed, it is found that these two advanced-fuel plants produce a larger net electrical power than is possible with a D-T F RTP designed under the same ground rules. This occurs because the neutron wall-loading limit requires a relatively low magnetic field (~25kG), hence low power density. If appropriate methods are found (e.g., a disposable or liquid-metal first wall) to handle higher neutron loadings, much higher power densities could be achieved in a D-T device by using higher fields. However, the added complexity of such designs would have to be compared with the relative simplicity of the advanced-fuel plants, especially their relatively long first-wall life and the non-breeding tritium blanket. While these results are quite preliminary, they serve to demonstrate that the proposed F RTP dynamic burn experiment holds the promise of scale-up to relatively attractive advanced fuel power plants.

Table IV Operating Parameters for Advanced-Fuel F RTP Reactors

Parameters	Cat-D	D- ³ He
External B_{ext} (kG)	86	100.
Peak electron density, $n(10^{21}\text{m}^{-3})$	1.9	2.9
Ion temp., T_i (keV)	40	50
Line density, $N_L(10^{21}\text{m}^{-1})$	5.6	7.0
Confinement time, τ_p (s)	7.0	8.0
Injection power P_{in} (MW)	26	38
Fusion power P_F (MW)	62	150
Power density, P_D (MW/m ³)	19	39
Neut./rad. wall loading		
$P_{\text{NW}}/P_{\text{RW}}$ (MW/m ²)	1.3/0.9	0.1/1.4
Q-value,	5.4	11
Net elec. output, P_{net} (MW)	27	69

8. Summary

The main point to be made from this study is that, based on present understanding of F RTP scaling, a high- β compact tori ignition (or burn) experiment appears feasible. A key aspect of the present design is the use of neutral-beam and pellet injection to supplement the implosion heating. There are alternate approaches, e.g., compression heating that might be considered. However, the neutral beam concept has the advantage of using proven technology while providing the dual functions of heating and rotation control. Additional experiments and theoretical studies are obviously necessary to verify the feasibility of this concept. However, due to the relative small size of the device, such experiments could be done with minimum cost and time. If successful, the high β of the device should open the door to attractive reactors including advanced-fuel Cat-D and D-³He devices.

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Figure Captions

Figure 1. Geometry of FRTP configuration.

Figure 2. Plausible particle paths leading to axial losses. Paths B, C and E are most probable, while Paths A, D and F rarely occur due to the higher energy required.

Figure 3. Magnetic field structure in the r - z plane. In a future D-T pilot unit an appropriate first-wall, blanket, and shield plus divertor "energy dumps" would be added.

Figure 4. Confinement regions: a) in H - P_θ space, and b) in velocity space for a specific radial position $0 < r < r_w$. Losses occur by scattering and transport across the spherical surface in (b) which, in (a) corresponds to leakage across the straight line representing the absolute confinement boundary.

Figure 5. $(n\tau_E)_{OP}$ and $(n\tau_E)_{PL}$ versus T_i for a D-T system. Here $(n\tau_E)_{OP}$ is based on LCST scaling while $(n\tau)_{PL}$ corresponds to energy breakeven.

Figure 6. An assumed ion temperature history for the FRTP burn experiment.