

NATIONAL INSTITUTE FOR FUSION SCIENCE

Investigation of D-³He Fusion Ignition in a Field-Reversed Configuration Modeled as a Highly Elongated Spherical Tokamak

O. Mitarai, T. Asai, M. Sakamoto, and Y. Kawamoto

(Received – Nov. 15, 2025)

NIFS-1132

April 6, 2026

This report was prepared as a preprint of work performed as a collaboration research of the National Institute for Fusion Science (NIFS) of Japan. The views presented here are solely those of the authors. This document is intended for information only and may be published in a journal after some rearrangement of its contents in the future.

Inquiries about copyright should be addressed to the NIFS Library, National Institute for Fusion Science, 322-6 Oroshi-cho, Toki-shi, Gifu-ken 509-5292 JAPAN.

E-mail: toshonifs@nifs.ac.jp

<Notice about copyright>

NIFS authorized Japan Academic Association For Copyright Clearance (JAC) to license our reproduction rights and reuse rights of copyrighted works. If you wish to obtain permissions of these rights, please refer to the homepage of JAC (<http://jaacc.org/eng/>) and confirm appropriate organizations to request permission.

Investigation of D-³He Fusion Ignition in a Field-Reversed Configuration Modeled as a Highly Elongated Spherical Tokamak

O. Mitarai ⁽¹⁾, T. Asai^(2,3), M. Sakamoto^(3,4), and Y. Kawamoto⁽⁵⁾

(1) Tokai University, Professor Emeritus, Kumamoto, Japan

(2) Nihon University, Tokyo, Japan

(3) LINEA Inovations, Inc. Tokyo, Japan

(4) Tsukuba University, Tsukuba, Japan

(5) National Institute for Fusion Science, Toki, Japan

Abstract

Ignition of the D-³He Field-Reversed Configuration (FRC) has been studied by assuming FRC as an extremely elongated spherical tokamak (ST) without the toroidal magnetic field. This assumption is based on the fact that the cross-sectional shape of an FRC closely resembles that of a highly elongated ST. Although confinement scaling is not well established for FRC, a tokamak energy confinement scaling law is well known. Given that a reliable confinement scaling law for FRC has not yet been established, we have applied a well-known tokamak scaling law to estimate ignition conditions. With the help of tokamak scaling, we can study an outline of ignition access and gain insight on what we should do next for achieving a D-³He FRC reactor.

Especially, we have proposed to employ the vertical (axial) magnetic field to induce the plasma current in FRC as used in a tokamak and ST. The gradual increase in plasma current up to ~ 200 MA with the help of 1.5 MeV NBI injection (300 - 400 MW) can bring an ignition with 2 GW fusion power. Energy conversion from the bremsstrahlung power and the divertor heat flux, and 14 MeV protons lost from the end mirror could provide 1.2 GW electric power output. However, it was pointed out that the present experimental data on the tilting stabilization are not enough, then more experimental and theoretical efforts are required to achieve a D-³He FRC reactor.

Keywords : D-³He fusion, FRC, ignition, spherical tokamak, plasma current startup

1 Introduction

Since the late 1980s (for ~ 36 years), one of authors (O.M.) has been studying on D-³He fusion reactors, especially for tokamak [1].

1. High aspect ratio tokamak has a large toroidal field, leading to the large synchrotron radiation loss. Wall reflectivity should be larger than 90% ($R_{\text{eff}} > 0.9$) to reduce the synchrotron radiation loss. This could be problematic due to first wall contamination by impurities. The plasma current should be more than 50 MA, which could damage the machine if the disruption takes place once. For He and proton ash exhaust, the ash confinement time should be comparable to

the energy confinement time ($\tau_{\text{ash}}/\tau_E \sim 1$) for D-³He ignition. Thus we judge that its feasibility is quite low [2].

2. Spherical tokamak has a lower toroidal field and higher beta, leading to the smaller synchrotron radiation loss. Therefore, the wall reflectivity can be lower as $R_{\text{eff}} \sim 0.4$ [2]. However, the plasma current should be ~ 100 MA which is larger than the high aspect ratio tokamak. This may be more severe than the high aspect ratio tokamak in terms of disruption. Although for He and proton exhaust, somewhat relaxed condition $\tau_{\text{ash}}/\tau_E \sim 2$ is needed for D-³He ignition, we judge its feasibility low [3].
3. In Reversed Field Pinch (RFP) reactor, D-T to D-³He Ohmic ignition is possible for $I_p \sim 60$ MA for the confinement enhancement factor of 2 over the IPB98y2 scaling [1]. We believe that disruptions may be avoided with active coils. However, active coils should be placed on the vacuum chamber to maintain the close proximity. For that the high temperature resistant coils should be developed, which is beyond our imagination. Fueling and He exhaust is difficult to achieve because RFP plasma is highly non-linear. We have decided to discontinue further consideration of this reactor concept for the time being.
4. Helical system has no plasma current in principle, then current disruption could be avoided. Therefore, D-T fusion reactor with safer operation is possible. He exhausts could be possible using the magnetic islands and fueling by pellet injection could be partly possible due to lower operating plasma temperature than tokamak reactor [4][7]. However, D-³He fusion is very difficult due to the inferior confinement to that of a tokamak. Because large confinement enhancement factor is required to reach ignition [8].

As one of authors (OM) cannot provide clear solution on above problems at present, one of authors has given up on continuing D-³He tokamak fusion reactor studies since 2022. Although FRC concept has not been in the scope of our studies for the past 36 years because the good experimental data aiming at a fusion reactor have not been obtained yet. However, recent TAE experiments have provided favorable experimental data with the relatively long stable discharge depending on the NBI injection period and large ion to electron temperature ratio [9]-[14]. Also tenfold confinement enhancement over the traditional FRC confinement time scaling has been obtained [9]. Furthermore, in the compressed FRC device, 1 keV electron temperature and 8 keV ion temperature have been observed [15]. Thus, recent development in FRC stimulates our reactor studies.

On the other hand, we notice that FRC can be regarded as a limiting case of a spherical tokamak (ST) with vanishing toroidal magnetic field, as suggested by their similar highly elongated cross-

sectional shapes and the presence of a toroidal plasma current without an externally applied toroidal field (Fig. 1 in Reference [16]). We also show the calculation result of the Hill's vortex for ST without the central solenoid, which can be considered to be FRC without toroidal field as shown in Fig. 1, where (1) torus with the toroidal plasma current in FRC corresponds to ST without the toroidal field, and (2) surrounding area without the plasma current in FRC corresponds to the scrape off layer in ST.

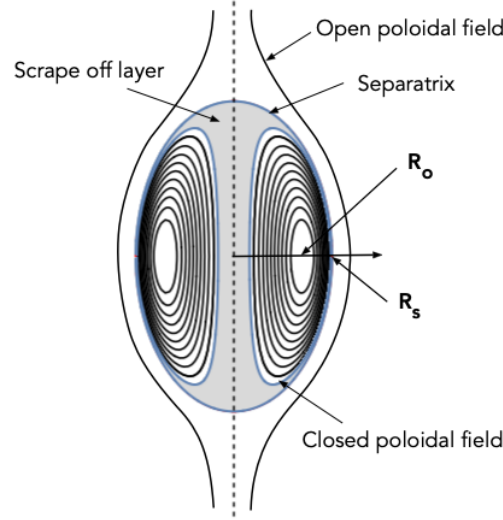


Fig.1 Hill's vortex to express magnetic surface in an ST and FRC

This idea has been also stimulated from our old paper on the vertical field formula for ST. "We note that when the aspect ratio approaches unity in an ST, its the vertical field formula reduces to that for an FRC as shown in Eq. (2.18)" in Reference [17]. This sentence inspired one of authors (OM) to use D-³He ST computer program for D-³He FRC reactor studies. In this study, we build upon previous studies on advanced fuel reactors and apply the ST code to explore D-³He FRC thermal fusion reactor [3]. Our approach is as follows.

(1) In ST program, the major radius is gradually reduced from 7 m to 2.6 m, and elongation is expanded from $\kappa = 3$ to 8 - 10, approaching FRC reactor.

(2) As the confinement scaling, we utilize the IPB98y2 scaling [18] which was used for ST reactor studies. Here, we use the mixed scaling with Neo-ALCATOR scaling (τ_{NA}) [19] to avoid the infinitely large value in the initial lower heating power phase. The minimum value given by $\tau_E = \text{Min}\{\tau_{NA}, \tau_{HH}, \tau_{IPB98y2}\}$ is used upon comparing two confinement times.

(3) As no toroidal field in FRC, we use $B_t = 1\text{T}$ to remove its effect from the scaling expression. If we put $B_t = 0$ into this scaling, the confinement time becomes zero, which contradicts the actual situation. Here we use

$$\tau_{IPB(y,2)} = \gamma_{HH} 0.0562 A_i^{0.19} I_p^{0.93} [MA] n_{19}^{-0.41} [\times 10^{19} m^{-3}] R_o^{1.97} [m] \epsilon^{0.58} \kappa^{0.78} B_{to}^{0.15} [T] / P_{NET}^{0.69} [MW] \quad (1)$$

$$\tau_{NA}(s) = 7.1 \times 10^{-22} \bar{n} [\text{cm}^{-3}] R^{2.04} [\text{cm}] a^{1.04} [\text{cm}] \sqrt{q_a}$$

(4) As the poloidal field exists only near the plasma surface, the synchrotron radiation is assumed to be zero.

(5) Other important ST parameters are changed to FRC parameters, such as to Spitzer resistivity for the plasma resistance, NBI current drive scheme from ST to FRC configuration, and BS current in an ST to a diamagnetic current in FRC.

Those parameters are adopted from ST studies, assuming that it is not strongly affected by FRC-specific features such as high β or lack of toroidal field. Since an FRC can be regarded as the high-elongation limit of a spherical tokamak, the value is expected to be approximately valid and is thus provisionally applied in this study. We have also intentionally adopted optimistic assumptions for the confinement time scaling in order to clarify the conditions required for D-³He ignition and to identify the key research areas that should be addressed after ignition is achieved. Because the purpose of this paper is to show what we should do to reach ignition and what should be studied after reaching ignition.

In Section 2, previous steady state FRC reactor designs are briefly summarized. In Section 3, formulae for D-³He fusion ignition calculation are described. In Section 4, ignition accesses by NBI current drive and ignition with 14 MeV proton loss for energy conversion are presented for various parameters, and ignition shutdown process is shown for safer operations. In Section 5, the tilting stability parameters during ignition access are examined. In Section 6, discussions on further issue are presented. Section 7 provides summary.

2 Previous steady state FRC reactor design

The most well-known FRC reactor design is ARTEMIS proposed by Prof. Momota [20]-[22]. The plasma diameter was $d = 4.0$ m, the plasma length was $L = 30$ m, and the plasma volume is $V = 376 \text{ m}^3$ as shown in Fig. 2(a). At the both mirror ends of the FRC, two traveling wave energy convertors are installed for energy conversion from lost 14 MeV protons. Another example of D-³He FRC was presented by Kernibecherer [22]. In contrast, D-T FRC reactor with 8.6 m plasma diameter, 40 m length, and 2000 m^3 plasma volume was given by Prof. Hirano [23] as shown in Fig. 2(b).

Recent design is a pulsed D-T FRC reactor by compression of plasma to increase the density and temperature. Therefore, the axial length is shorter as ~ 12 m and the reactor size is smaller than above two steady state devices [24].

We propose a feasible D-³He FRC reactor with $d = 10$ m plasma diameter, $L = 40$ m length, and $2000 \sim 2500$ m³ plasma volume in this work. We assume the elongation of $E = 4$ to suppress the tilting instability as shown in Fig. 2(c). Most distinguishing feature of this proposed reactor design is using the large plasma current ~ 200 MA, which can be induced by time derivative of the axial magnetic field.

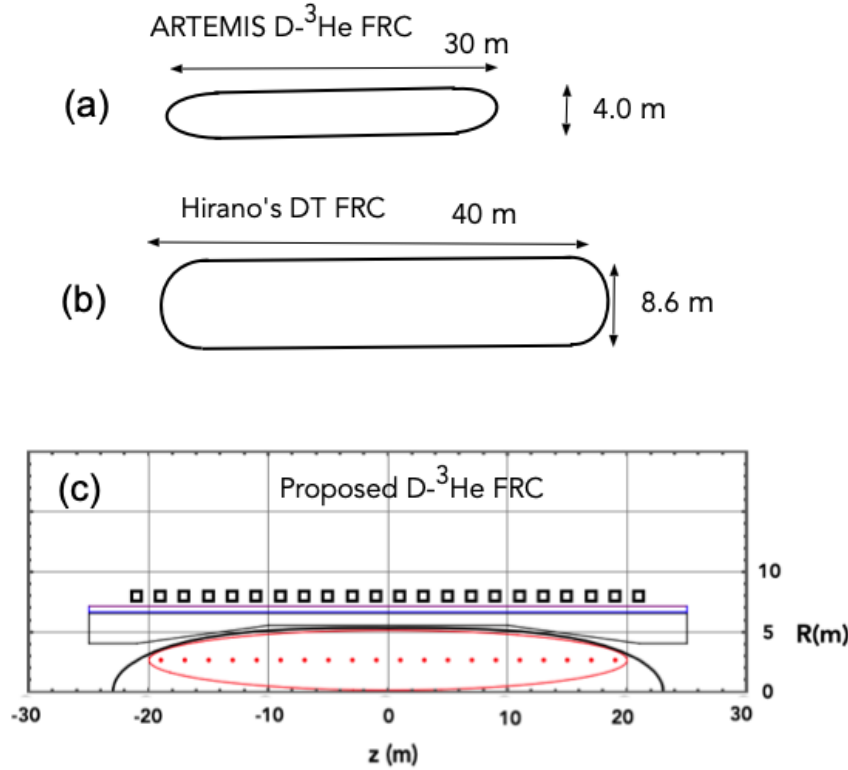


Fig. 2 Schematic drawing of D-³He FRC reactor ARTEMIS, D-T-FRC and the upper part of the proposed D-³He FRC reactor (Squares show solenoid, and red dots show the equally distributed plasma current)

We have investigated the device parameters for D-³He FRC reactor, starting from ST to FRC size. We have gradually changed the machine parameters from $R = 7$ m, 6 m, ... to 2.6 m capable of ignition using the mixed (ST and FRC) method initially. This size transition is relatively easy due to the large plasma current induction effect by the vertical field, which depends on the plasma size. The current drive and BS current effect is small in a D-³He fusion due to the very high temperature. For this purpose, many modified computational codes were developed to accommodate the parameter variations. Finally, all the computational parameters are converted to FRC's ones.

Final device size was determined as follows:

- (1) $R = 2.6$ m, $a = 2.5$ m, $\kappa = 8.0$ (for ST) or $E = 4$ (for FRC), $L = 40$ m, $V = 2566$ m³
- (2) $R = 3.0$ m, $a = 2.1$ m, $\kappa = 9.8$ (for ST) or $E = 4$ (for FRC), $L = 40$ m, $V = 2559$ m³

Here, the definition of the elongation is given by $\kappa = (L/2)/a$ for a ST, and $E = (L/2)/(2a)$ for a FRC.

For NBI (1~2 MeV) injection, the density should be at most $2 \times 10^{20} \text{ m}^{-3}$. As smaller size devices than this have higher density for ignition, we employ above size to inject NBI.

3 Formula for D-³He ignition calculation

We have utilized computer programs developed for ST research, consisting of the following equations [3] [25]. Each parameter has been changed from ST to FRC configuration.

3.1 Fuel particle balance equations for D and ³He

In this reactor, as main fusion reactions come from the reaction of $\text{D} + {}^3\text{He} \rightarrow {}^4\text{He} (3.67 \text{ MeV}) + \text{p} (14.67 \text{ MeV})$, fueling of D and ³He is necessary. D and ³He fueling particle confinement time to the energy confinement time ratio are tentatively assumed as $\tau_{\text{D}}/\tau_{\text{E}} = 1.3$ and $\tau_{\text{He}}/\tau_{\text{E}} = 1.0$ (Note: this is the same value with the ash particles), respectively. As these values affect the fueling method, its effect should be studied in more detail together with fueling studies. Initially D:³He fuel ratio is set to 2:1 to maximize the fusion reactivity, and then changed to ³He rich operation of 0.45:0.55 at 250 s during the NBI heating phase to reduce the neutron production from the D-D reaction, while maintaining sufficient fusion reactivity for sustaining the burn. These parameters are taken into fuel particle balance equations.

3.2 Ash particle balance equations for ⁴He and 14 MeV proton ash

D-³He fusion reactions provide ash particles such as ⁴He, p and D-D side fusion reactions do ³He, T and p, and ⁴He particles as shown in the reactions: $\text{D} + \text{D} \rightarrow {}^3\text{He}(0.82 \text{ MeV}) + \text{n}(2.47 \text{ MeV})$, $\text{D} + \text{D} \rightarrow \text{T}(1.01 \text{ MeV}) + \text{p}(3.03 \text{ MeV})$, and $\text{D} + \text{T} \rightarrow {}^4\text{He}(3.52 \text{ MeV}) + {}^1\text{n}(14.05 \text{ MeV})$. For simplicity, we assume that the ash confinement time to the energy confinement time ratio for all ash particles is set to $\tau_{\text{ash}}/\tau_{\text{E}} = 1$, removing ash as soon as possible after providing its energy to plasma. This is most important in an advanced fuel fusion for ignition. In D-³He ST design, we have assumed all the ash confinement time ratio is $\tau_{\text{ash}}/\tau_{\text{E}} = 2$, as large as possible for reaching ignition because its small value seems to be difficult to obtain in experiments. However, in this FRC study we assume $\tau_{\text{ash}}/\tau_{\text{E}} = 1$ from the outset to ensure ignition. This is because this study is aiming at search what should be done next after reaching ignition in FRC as mentioned already. Fortunately, as the particle and energy confinement time ratio of $\tau_{\text{particle}}/\tau_{\text{E}} = 1$ is reported [15], this assumption seems to be justified. These parameters are taken into ash particle balance equations.

3.3 Ion and electron, separate power balance equations

In an advanced fuel fusion, the hot ion mode is required to reach ignition. While the electron temperature is better to be low to reduce the bremsstrahlung loss and the synchrotron radiation loss,

the ion temperature should be high enough for large fusion reactivity by an efficient ion heating. In the bremsstrahlung loss power, the electron-ion and electron-electron scatterings are also taken into account with the relativistic effects [3][25]. As mentioned in Introduction, the synchrotron radiation is assumed to be zero because the poloidal field exists only near the plasma surface and the plasma beta is relatively high. For efficient ion heating, the nuclear elastic scattering effect is taken into account [26]. In the ion and electron power balance equations, the ion to electron confinement time ratio of $\tau_i/\tau_e \sim 15$ is chosen to ensure the hot ion mode of $T_i/T_e > 1.5$, as observed in C-2 experiments [11]-[13], though the physical mechanism for such high τ_i/τ_e remains to be investigated.

In this study the fusion product particles such as 14 MeV proton and 3.67 MeV ^4He are all confined 100 % to ensure the ignition. In the next step, to study the possibility of the energy conversion by the fusion product, the loss fraction of 14 MeV proton is introduced using the coefficient of “ETP14” as will be seen in Fig. 8 and later. As the coefficient (1-ETP14) is multiplied at 14.67 MeV term in the fusion power, the power balance will be affected.

3.4 Density and temperature profile

In the rigid rotor model (RRM) in FRC, the density profile is commonly given by [27]

$$n(r) = \frac{n(0)}{\cosh^2 \left[\frac{(r^2 - r_0^2)}{(r_0 \Delta r)} \right]} \quad (2)$$

where r_0 corresponds to the major radius R in a tokamak, and Δr to the minor radius a . However, as it may consume a long time to take this profile effect into account in this urgent ignition study, we have assumed the parabolic density and temperature profile as a first order approximation as used in ST studies,

$$n(r) = n(0) \left(1 - \left(\frac{r - r_0}{\Delta r} \right)^2 \right)^{\alpha_n}, \quad T(r) = T(0) \left(1 - \left(\frac{r - r_0}{\Delta r} \right)^2 \right)^{\alpha_T} \quad (3)$$

where α_n is the density profile factor and α_T is the temperature profile factor. Comparison FRC density profiles with the various parabolic profiles are shown in Fig. 3. FRC density profile is found to be steeper than the parabolic profile.

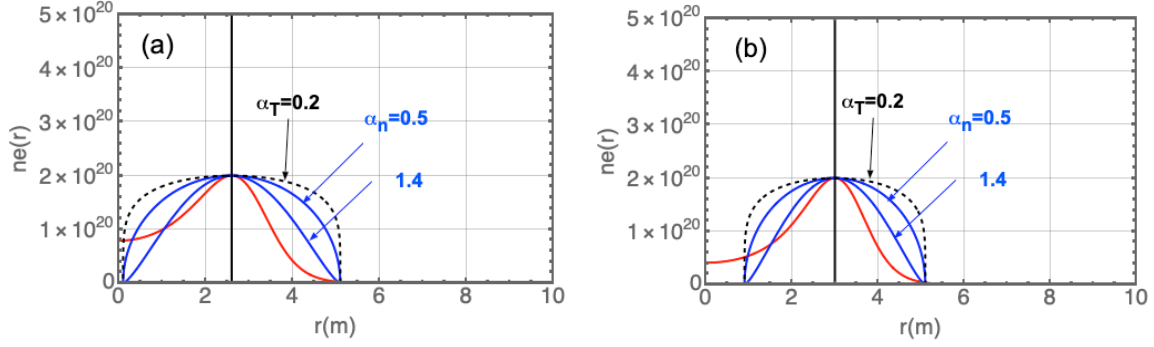


Fig. 3 (a) Density and temperature profile in the case of $r_0 = R = 2.6$, and $\Delta r = a = 2.5$. Red line shows the FRC density profile, blue lines the parabolic profile of $\alpha_n = 0.5, 1.4$, and dotted line, the temperature profile $\alpha_T = 0.2$
(b) Density and temperature profile in the case of $r_0 = R = 3.0$ and $\Delta r = a = 2.1$

3.5 Plasma circuit equation

In a magnetic confinement fusion device, an inductive operation is basic and valuable to induce the plasma current, which is based on the electromagnetism. This induction method is very useful in the tokamak operation because the density limit and huge current drive power can be avoided. One of authors (O.M.) demonstrated an alternating current (AC) operation in the STOR-1M tokamak [28], and then proposed AC tokamak reactor to obtain the quasi-continuous discharge instead of the current drive technique [29]. Here an inductive operation has been utilized maximally. This idea was only limited to an “inboard solenoid (central solenoid)” to induce the plasma current in a tokamak.

However, when I saw a theta pinch experiment around 1975 in Kyushu University, I have noticed that the “outboard solenoid” placed outside of a plasma can induce the plasma current as schematically shown in Fig. 4. The one turn cylindrical coil current can produce the plasma current to the opposite direction. In theta pinch experiments, the produced plasma current is compressed and does not last long because no external plasma heating. Therefore, if the heating power is continuously applied to the plasma to prevent the compression, the plasma current would be increased by induction effect of the axial magnetic field created by the one turn cylindrical coil.

When the theta pinch device (Fig. 4(a)) is rotated 90 degrees to stand vertically (Fig. 4(b)), and then if the outer cylindrical coil is split up into two coils, that seems to work as a vertical field coil in a tokamak or ST (Fig. 4(c)). Thus, we understand that the vertical field coil can induce the plasma current together with heating power in an ST. “When a plasma is heated by NBI et al, the plasma expands to outboard side. To keep the plasma at the center, the vertical field is applied, leading to the plasma current ramp-up by flux input.” This idea was proposed in Reference by one of authors [30]. The very clear experimental result to verify this idea was found in Fig. 5 in JT60U paper [31], then we further analyze that experimental data to establish this concept [32]. Further

experimental observation is obtained in TST [33], and tried in UTST experiments [34], and finally demonstrated that the plasma current can be started up by the outer vertical field coil without the central solenoid in the iron core STOR-M tokamak [35]-[37]. Thus, the plasma current can be ramped-up by the vertical field B_v in the tokamak and ST, and then this idea is re-imported into FRC in this study (Fig. 4(d)).

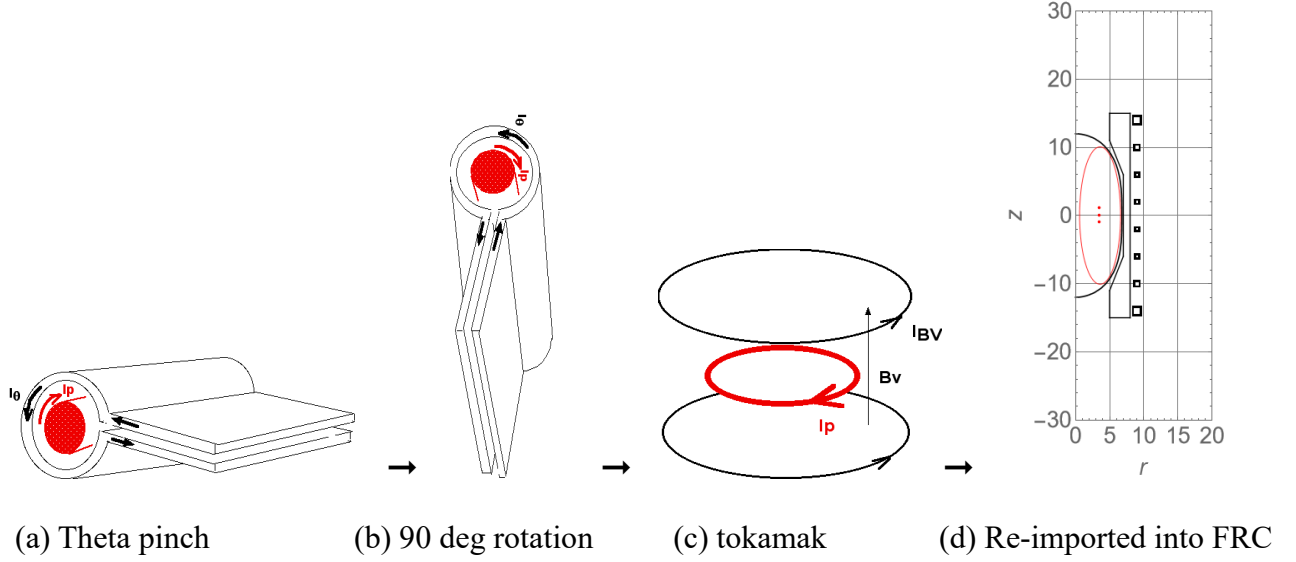


Fig. 4 The transition of conceptual idea of the plasma current ramp-up from theta pinch to FRC via tokamak and ST

The circuit equation is composed of the plasma self inductance, plasma resistance, and voltage induction term. The plasma inductance for ST was given by Hirshman and Neilson [38] as shown here.

$$L_p = \mu_o R \left\{ \frac{a(\varepsilon)(1-\varepsilon)}{(1-\varepsilon) + b(\varepsilon)\kappa} + \frac{\ell_i}{2} \right\} \quad (4)$$

where the first term is the external inductance, ε is the inverse aspect ratio, κ is the elongation, ℓ_i is the internal inductance, and

$$a(\varepsilon) = (1 + 1.81\sqrt{\varepsilon} + 2.05\varepsilon) \ln(8/\varepsilon) - \{2.0 + 9.25\sqrt{\varepsilon} - 1.21\varepsilon\}$$

$$b(\varepsilon) = 0.73\sqrt{\varepsilon} \{1 + 2\varepsilon^4 - 6\varepsilon^5 + 3.7\varepsilon^6\}$$

$$c(\varepsilon) = \left\{ \left(\frac{0.905}{\sqrt{\varepsilon}} + 2.05 \right) \ln(8/\varepsilon) - \left(0.84 + \frac{1}{\varepsilon} + \frac{6.435}{\sqrt{\varepsilon}} \right) \right\} (1-\varepsilon) - a(\varepsilon)$$

$$d(\varepsilon) = -1 + \frac{0.73}{\sqrt{\varepsilon}} \{0.5 + 9\varepsilon^4 - 33\varepsilon^5 + 24.05\varepsilon^6\} \kappa$$

The loop voltage to increase the plasma current comes from the vertical field in ST, which is derived from the plasma inductance [17] as

$$B_{VE} = \frac{\mu_o I_p}{4\pi R} \left\{ \frac{1}{\mu_o} \frac{\partial L_p}{\partial R} + \left(\beta_p - \frac{1}{2} \right) \right\} = \frac{\mu_o I_p}{4\pi R} \left\{ \frac{1}{\mu_o} \frac{\partial L_s}{\partial R} + \frac{\ell_i}{2} + \beta_p - \frac{1}{2} \right\} \quad (5)$$

where L_s is the external inductance and

$$\frac{\partial L_s}{\partial R} = \mu_o \left\{ \frac{a(\varepsilon)(1-\varepsilon)}{(1-\varepsilon)+b(\varepsilon)k} \right\} - \mu_o \frac{\varepsilon c(\varepsilon)}{(1-\varepsilon)+b(\varepsilon)k} + \mu_o \left\{ \frac{a(\varepsilon)(1-\varepsilon)}{(1-\varepsilon)+b(\varepsilon)k} \right\} \frac{\varepsilon d(\varepsilon)}{(1-\varepsilon)+b(\varepsilon)k}$$

Using Eq.(4) and (5), the plasma circuit equation can be obtained as

$$L_p \frac{dI_p}{dt} + \overline{R_p} (1 - F_{CD} - F_{BS}) = - \frac{d(M_{pV} I_V)}{dt} = - (\pi R_o^2) \frac{dB_{VE}}{dt} \quad (6)$$

where I_p is the plasma current, R_p is the volume averaged plasma resistance consisting of the spitzer resistivity, F_{CD} is the current drive fraction by NBI to the plasma current, and F_{BS} is the bootstrap current fraction for ST and the diamagnetic current fraction for FRC. The induction term is eventually expressed by the time derivative of the vertical field flux $\pi R_o^2 B_{VE}$ given by the solenoid. Therefore, when the plasma is heated, β_p is increased in Eq. (5) and the vertical field is increased, inducing the plasma current. As the plasma resistance is very small in high-temperature D-³He plasmas, Eq. (6) reduces to a flux balance relation, $L_p I_p = (\pi R_o^2) B_{VE}$, where the vertical field flux acts as the driving mechanism for plasma current. We note that I_p and β_p are taken as an averaged value at two consecutive computation times, otherwise numerical instability takes place in the initial phase.

3.5.1 The perpendicular resistivity in FRC

As the plasma current perpendicular to the poloidal field in FRC and D and ³He mixed ions yield a factor of 2 and 1.5 larger, respectively, we use tentatively a factor of 3.0 large Spitzer resistivity η_{sp} for calculating the plasma resistance. Thus, the volume averaged plasma resistance is simply calculated by

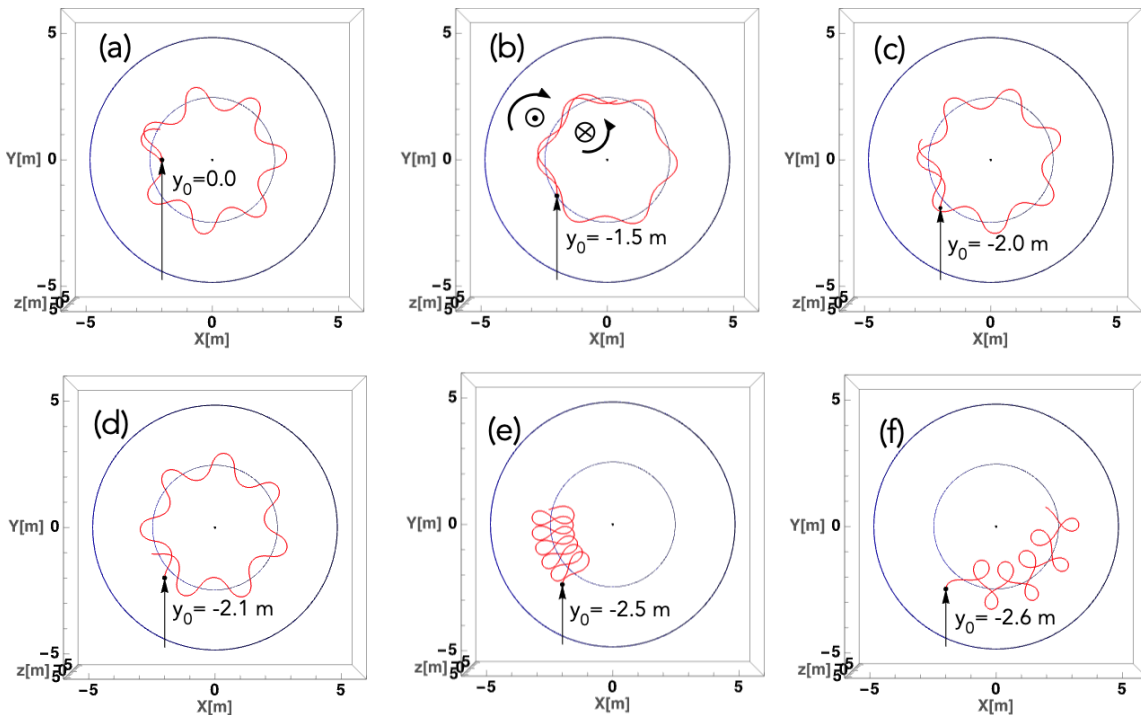
$$\overline{R_p} = 3\overline{\eta_{sp}} \frac{2\pi R}{\pi a^2 \kappa} \quad (7)$$

with the ST elongation κ , which provides the Ohmic heating power. However, as the plasma temperature is very high in a D-³He fusion plasma, the plasma resistivity itself is very small, which does not affect overall plasma behaviour. Also as the plasma current may flow near a magnetic field null in FRC, the plasma resistance may be much lower this value.

3.5.2 NBI current drive for FRC and betatron orbits

In the spherical tokamak, NBI is injected along the strong toroidal magnetic field line which is at the right angle to the poloidal field. Therefore, the injected ions simply move along the toroidal magnetic field line with Larmor motion. However, in FRC, while NBI is also injected along the toroidal direction, but due to no toroidal magnetic field its moving is constrained from the Lorentz force of $v \times B_p$ by the poloidal field. Thus, NBI is injected around the null point of the poloidal field. As a result, beam ion trajectory takes various shapes (called betatron orbit) depending on the initial NBI ionized point as shown in Fig. 5 [27] [39] [40].

For example, we show the various betatron orbits depending on the NBI ionization points at $t = 50$ s during NBI injection phase accessing ignition in Fig. 18 with $\alpha_n = 0.5$ and $\alpha_r = 0.5$. In this calculation, the simplest model is employed for the Lorentz force exerted on beam ion on the flat plane. NBI injection point is along the straight line at $x_0 = -2.1$ m, and y_0 positions were changed for 0 to -2.6 m as shown in Fig. 5. When the NBI is ionized at the place of $y_0 > -2.5$ m as shown in (a) to (e) in Fig. 5, NBI ion can circulate clockwise direction, which provide the driven current. However, for the NBI ionized position of $y_0 < -2.5$ m as shown in (f) in Fig. 5, NBI ions circulate in opposite direction, reducing the current drive capability. We see the various betatron orbits depending on the ionization points. Inside the magnetic field null circle, ion motion is anti-clockwise direction, and in the outside of it ion motion is clockwise direction as shown in Fig. 5(b). Therefore, when ion crosses the null point circle, ion moves clockwise and anticlockwise direction alternatingly, leading to the Betatron orbit.



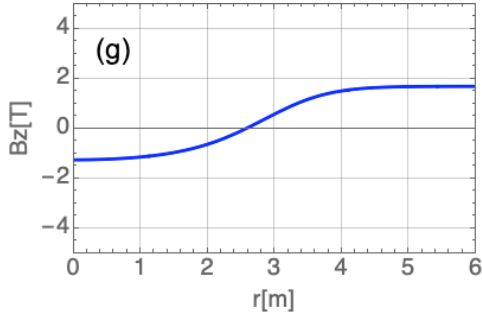


Fig. 5 (a)-(f) Various betatron orbits (red curves) depending on the position of NBI ionization points for $y_0 = -0.0 \text{ m} \sim -2.6 \text{ m}$ at $x = -2.1 \text{ m}$. Inner circle shows the null point at $r = 2.6 \text{ m}$. The magnetic field direction and the rotating direction of the NBI ion are shown in (b). (g) Magnetic field profile at $t = 50 \text{ s}$ in Fig. 8 with $R = 2.6 \text{ m}$, $a = 2.5 \text{ m}$, $\alpha_n = \alpha_T = 0.5$ and $P_{\text{ext}} = 380 \text{ MW}$.

As the NBI ion orbit depends on the ionized point along the injection path, we estimate the injected NBI driven current in the best injection case such as $(x_0, y_0) = (-2.6 \text{ m}, 0 \text{ m})$, which provides the maximum value. Beam ions with injection beam velocity U_{NBI}

$$U_{\text{NBI}} = 1.3838 \times 10^4 \sqrt{\frac{E_{\text{NBI}} (eV)}{A_b}}$$

can go around the circumference length of $2\pi R$ during the beam slowing down time τ_{SD} . [41]. The rotation number of beam ion is given by $U_{\text{NBI}}\tau_{\text{SD}}/(2\pi R)$, and the current emitted from NBI system is I_{NBI} , and the injection angle to the equatorial plane of FRC is θ_{deg} , (Note: in this study θ_{deg} is set at $20 \sim 30$ degrees), the NBI driven current is calculated by

$$I_{\text{CD}} (A) = \left(\frac{U_{\text{NBI}} \tau_{\text{SD}}}{2\pi R} \right) I_{\text{NBI}} \cos \theta_{\text{deg}}$$

As the emitted NBI current I_{NBI} is given by the NBI total power,

$$P_{\text{NBI}} (MW) = I_{\text{NBI}} V_{\text{NBI}} (MV) \quad (8)$$

where V_{NBI} is the applied voltage in NBI system, the NBI driven current is finally given by

$$I_{\text{CD}} (A) = \left(\frac{U_{\text{NBI}} \tau_{\text{SD}}}{2\pi R} \right) \left(\frac{P_{\text{NBI}} (MW)}{V_{\text{NBI}} (MeV)} \right) \cos \theta_{\text{deg}} \quad (9)$$

Here, slowing down time is given by

$$\tau_{\text{slowing}} = \frac{\tau_{ie}(r)}{3} \ln \left(1 + \left(\frac{E_{\text{NBI}}}{\mathcal{E}_{cr}} \right)^{1.5} \right),$$

where $\mathcal{E}_{cr}$ is the critical energy

$$\varepsilon_{cr} = 14.8 \frac{A_b z_{eff}^{2/3}}{A_i^{2/3}} T_e = 18.6 T_e (keV)$$

$$\tau_{ie} = \frac{3.16042 \times 10^{-6} A_b T_e (eV)^{1.5}}{Z^2 (n / 10^{20}) \log \Lambda}$$

For example, for the D-beam parameter such as $E_{NBI} = 1.5$ MeV, $Z = 1$, $T_e = 50$ keV, $R_0 = 2.6$ m, $\theta = 20$ degree, the NBI driven current is given by $I_{CD} = 194$ MA for $P_{NB} = 380$ MW, $V_{NBI} = 1.5$ MV, $I_{NBI} = 253$ A, $n_e = 1 \times 10^{20} \text{ m}^{-3}$, $\tau_{ie} = 8.11$ s and $\tau_{sd} = 1.4$ s. This example parameter almost corresponds to $t = 100$ s in Fig. 8. In this FRC system we consider 20 NBI injectors with each power of 25 MW.

3.5.3 Magnetic field profile for NBI injection and diamagnetic current

The magnetic field profile and its null point are important for NBI injection and diamagnetic current estimation in FRC. This magnetic field profile comes from the rigid rotor model (RRM) [27]. But its model relies on the short pulse discharge such as theta pinch experiments. The magnetic field produced by the plasma current in the cylindrical tube is confined inside the metal tube because of its short discharge length. This may be applicable to short-pulse compressed FRC experiments [24].

However, in a FRC aiming at the steady state operation, the magnetic field diffuses outside the metal tube and the solenoid coil. Therefore, traditional RRM needs to be modified. Here we calculate the magnetic field profile using the elliptic integral method as used in tokamak studies.

3.5.3.1 Magnetic field profile in RRM with short pulse model

In this model the same magnetic field is assumed inside and outside of the solenoidal shape of the injected ion beam current I_θ , corresponding to the plasma current [27]. The boundary condition inside the plasma current is given by $B_z(0) = -B_o + \mu_o I_\theta / 2$ and the one outside the plasma current is $B_z(\sqrt{2}r_o) = -B_o - \mu_o I_\theta / 2$, respectively, which provides the magnetic field profile.

$$B_z(r) = -B_o \left[1 + \frac{\mu_o I_\theta}{2B_o} \tanh \left(\frac{r^2 - r_o^2}{r_o \Delta r} \right) \right] = -B_o \left[1 + \sqrt{\beta} \tanh \left(\frac{r^2 - r_o^2}{r_o \Delta r} \right) \right] \quad (10)$$

where the β is the plasma beta value, r_o corresponds to the major radius, Δr corresponds to the minor radius and B_o is the external magnetic field. Here, we note that the dimension of I_θ is the beam current per unit length [A/m].

3.5.3.2 Proposed steady state magnetic field profile

In general, leakage magnetic field from a solenoid coil at the steady state condition is almost zero. Therefore, the boundary condition inside the plasma is the same as $B_z(0) = -B_o + \mu_o I_\theta / 2$ and the one outside of plasma $B_z(\sqrt{2}r_o) = -B_o + 0$. Eq.(10) is heuristically modified to satisfy above boundary condition keeping the same profile shape. This modified B_z profile corresponding to the steady state condition is finally calibrated by the elliptic integral method. The procedures are described here.

(1) To remove the magnetic field outside the solenoid, we make the total magnetic field half at first.

$$B_z(r) / B_o = 0.5(\mu_o I_\theta [A / m] / 2B_o) \tanh\left(\frac{r^2 - r_o^2}{r_o \Delta r}\right)$$

And then by subtraction of "1" from "tanh" term, zero magnetic field is obtained with keeping the same profile,

$$B_z(r) / B_o = 0.5(\mu_o I_\theta [A / m] / 2B_o) \left\{ \tanh\left(\frac{r^2 - r_o^2}{r_o \Delta r}\right) - 1 \right\}$$

(2) "+1" is added to provide the outside magnetic field produced by the solenoid

$$B_z(r) / B_o = 1 + 0.5(\mu_o I_\theta [A / m] / 2B_o) \left\{ \tanh\left(\frac{r^2 - r_o^2}{r_o \Delta r}\right) - 1 \right\}$$

(3) Final magnetic field profile is given by

$$B_z(r) = B_o + 0.5(\mu_o I_\theta [A / m] / 2) \left\{ \tanh\left(\frac{r^2 - r_o^2}{r_o \Delta r}\right) - 1 \right\} \quad (11)$$

When the plasma current in FRC is increased, the magnetic field decreases inside the plasma, leading to the reversed magnetic field. The coefficient $(\mu_o I_\theta / 2)$ is proportional to the induced plasma current. We determine this coefficient using the steady state magnetic field produced by the distributed solenoidal plasma current as indicated by red dots in Fig. 2(c) with the elliptic integral method as tokamak studies (blue solid line in Fig. 6). Thus, the final formula for the calibrated magnetic field profile at $z = 0$ produced by the total plasma current I_p [A] is given by

$$B_z(r) = B_v + 1.12 \times 0.5 \times C \times I_p [A] \left\{ \tanh\left(\frac{r^2 - r_o^2}{r_o \Delta r}\right) - 1 \right\} \quad (12)$$

where $C = +0.034542 \times 10^{-6}$ and the factor 1.12 is a fine tuning factor. This is shown by the black solid line in Fig. 6. We use this analytical expression to calculate the null position for NBI

injection and the estimation of the diamagnetic current. Now B_v is the axial magnetic field produced by the solenoid coil, and the coefficient of $1.12 \times 0.5 C I_p$ provides the reversed magnetic field produced by the plasma current as shown in Fig.6. When B_v is increased by solenoid action due to the plasma heating, the plasma current is induced and then the reversed field is created. Thus, the null point is located almost at the same position.

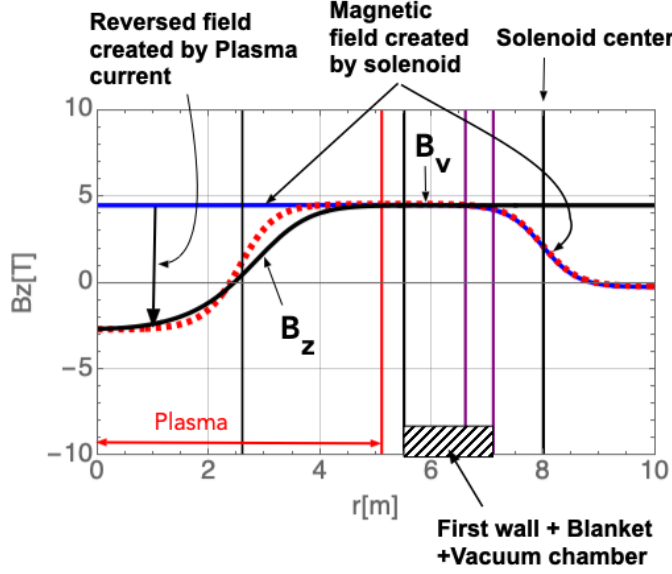


Fig. 6 Black solid line shows the reversed magnetic field profile B_z at $z = 0$ given by Eq.(11) for $r_o = 2.6$ m and $\Delta r = 2.5$ m. Blue solid line shows the steady state magnetic field profile created by the solenoid placed at $r = 8$ m which is calculated by the elliptic integral. Red dotted line shows the actual magnetic field in the plasma and the blanket space created by the solenoid. The plasma current $I_p = 200$ MA is placed at $r = 2.6$ m. We adjust both values at $r = 0$.

Eq. (12) also yields the FRC density profile as given by Eq. (2) through the relation of $\mu_o J_\theta = -\partial B_z / \partial r$ because the constant coefficient was only modified.

3.5.4 The diamagnetic current

In a spherical tokamak, the bootstrap current plays an important role for maintaining the discharge length. On the other hand, in FRC the diamagnetic current plays an important role. It is obtained from the MHD equilibrium condition of $J \times B = \nabla P$. In general, the Grad-Shafranov equation is solved and then the diamagnetic current is calculated using the flux function as $j_\phi = 2\pi r (\partial P / \partial \Psi)$ with $\partial \Psi = 2\pi r B_z \partial R$ [42]. However, in this study to obtain the results and see the overall tendency quickly, the current density is directly calculated by

$$j_{dia}(r) = \frac{\partial P(r) / \partial r}{B_z(r)} \quad (13)$$

where $B_z(r)$ is the axial magnetic field profile including the null point which is given by Eq. (12),

and $P(r)$ is the total pressure profile consisting of the ion and electron energy

$P(r) = k_B n(r)(T_i(r) + T_e(r))$ with k_B being the Boltzman constant. This corresponds to the one-dimensional analysis.

This current density should be calculated with the magnetic field profile with the singularity $B_z(r) = 0$. However, this singularity is removable because the numerator/denominator in Eq. (12) has 0/0 when approaching the singularity point (the L'Hopital's rule). When the pressure peak and the magnetic null point are the same on the r axis, this current density can be calculated.

By integrating the current density, the total diamagnetic current is given by

$$I_{dia} = \int j(r) L(r) dr = \int_{R-a}^{R+a} \frac{(dP/dr)}{B_z(r)} 2b \sqrt{1 - \frac{(r-R)^2}{a^2}} dr \quad (14)$$

on the elongated elliptic cross section as shown in Fig. 7(f).

$$\frac{(r-R)^2}{a^2} + \frac{y^2}{b^2} = 1$$

In Fig. 7, the pressure profile $P(r)$, its gradient dP/dr , the magnetic field profile $B_z(r)$ and its reversal $1/B_z(r)$ are shown. Finally, the current density profile $j(r)$ is shown in Fig. 7(e), where singularity point is removed.

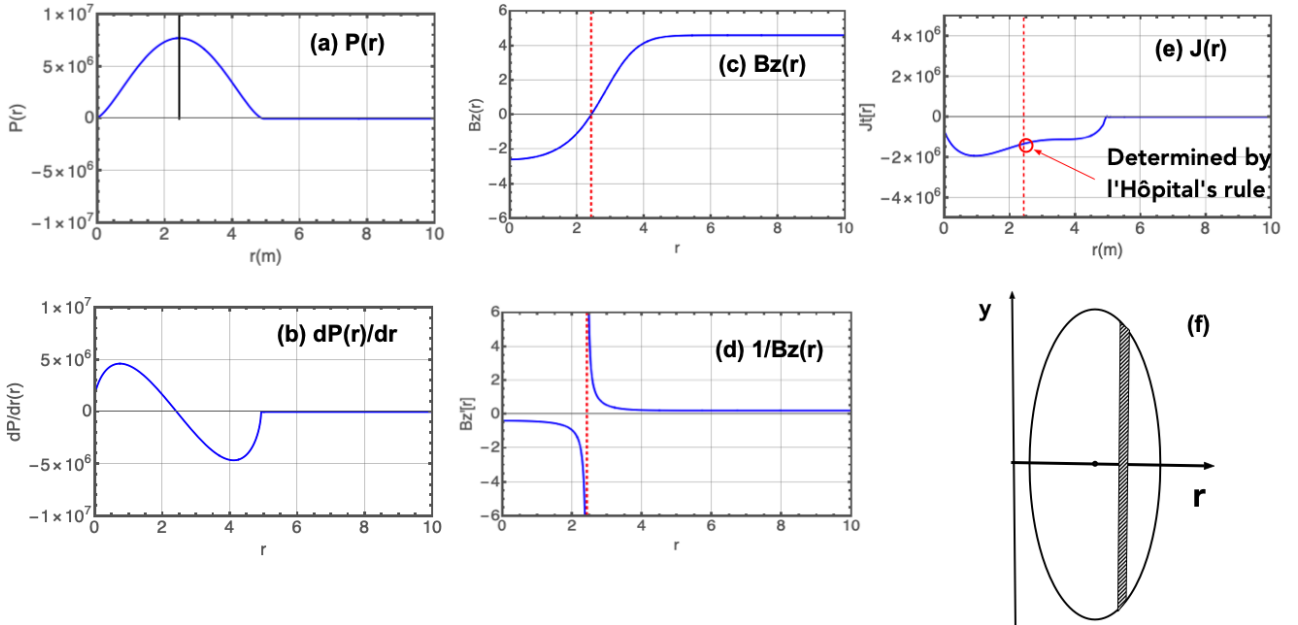


Fig. 7 (a) Plasma pressure $P(r)$, (b) gradient of the plasma pressure dP/dr , (c) magnetic field profile $B_z(r)$, (d) its reciprocal $1/B_z(r)$, (e) the calculated current density profile $j(r)$, and (f) elliptic cross section for calculation. Profiles are $\alpha_n = 1.4$ and $\alpha_T = 0.2$.

The temporal evolution of the diamagnetic current during the ignition access phase is evaluated in

the following section, based on the derived profile and current density formulation.

4. Ignition by NBI current drive in FRC

We should mention about the initial plasma state before applying NBI. We assume that the two CT rings are axially injected from the both sides as usually operated in FRC experiments in C-2 [43] [9], C-2U [10][11], C-2W [12]-[14], FAT-CM device [44]~[47]. After FRC plasma configuration and density are properly established for NBI injection, NBI is injected. In the following we consider discharge after initial FRC formation for two sizes of FRC device as (1) $R = 2.6$ m, $a = 2.5$ m and $\kappa = 8.0$, and (2) slightly larger major radius $R = 3.0$ m, $a = 2.1$ m and $\kappa = 9.8$ for accessing D-³He fusion ignition regime.

4.1. Ignition for broader density profile

As ignition capability depends on the density and temperature profiles, it is well known that it is more difficult to reach ignition with broader density and temperature profiles. Therefore, we first tried the relatively broader density and temperature profile of $\alpha_n = \alpha_T = 0.5$. As shown in Fig. 3, FRC density profile is much steeper than this profile, the difficult parameters have been chosen as a first trial to see the overall tendency. We show the calculation result on ignition for $R = 2.6$ m, $a = 2.5$ m and $\kappa = 8.0$, with the assumption of $\tau_{ash}/\tau_E = 1$ and the confinement enhancement factor of $\eta_{HH} = 1.3$, and fusion power set value is $P_f = 2$ GW at $t = 350$ sec in Fig. 8. When the external heating power of $P_{ext} = 380$ MW is applied for 200 s, the plasma current is increased to 180 MA with the vertical (axial) magnetic field and the driven NBI current is ~ 200 MA. The density and temperature are both increased, and then going to increase even after switching off the NBI at 200 sec until 350 s along the set value of the fusion power of 2 GW (Fig. 8(c)). We note the confinement time is governed by Neo-Alcator scaling (dashed line) during 0 $\sim$ 300 s as seen in Fig. 8(d). At steady state condition, the density and temperature are $n(0) = 1.65 \times 10^{20}$ m⁻³, $T_i = 178$ keV, $T_e = 107$ keV, $T_i/T_e = 1.66$, the plasma current reaches $I_p = 208$ MA, the axial magnetic field reaches $B_v = 4.5$ T, the confinement time is $\tau_E = 7.4$ s and the ⁴He and proton ash fraction is $f_{ash} = 3.6$ %. This ignition parameter of $n(0)\tau_E = 1.22 \times 10^{20}$ m⁻³s is in the range of $n(0)\tau_E = (1 \sim 4) \times 10^{20}$ m⁻³s for a D-³He tokamak reactor [48]. The total bremsstrahlung power is $P_B = 0.46$ GW, the plasma conduction loss to the divertor is $P_L = 1.5$ GW, the total neutron power is $P_n = 36$ MW and the total plasma energy is $W_p = 11.2$ GJ. The average beta value based on the axial magnetic field is 36 % and the peaked beta value is 72 %.

As the diamagnetic current separately calculated is also plotted in Fig. 8(f), the diamagnetic current fraction to the total plasma current is $f_{dia} = 60$ %. This provides the plasma current decay time of $\tau = L_p / \{R_p(1 - f_{dia})\} = 4.95 \times 10^{-7} / 1.98 \times 10^{-11} = 25,000$ s = 6.9 hours. Thus, steady state

operation cannot be obtained for the broader density profile. As this is a big problem to achieve a long pulse operation, we further study more peaked density profile in the next section.

If the magnetic field profile B_z is changed during NBI injection phase, it is difficult to inject NBI. Therefore, we have checked this problem. To see its effect quickly, it is convenient to calculate the I_p/B_v ratio. Because if it is changed, the null point is changed. As seen in Fig. 8(g), the I_p/B_v ratio is almost constant during NBI injection phase, only slightly decreasing. In the right column, the temporal change of the magnetic field profile given by B_v and plasma current I_p are shown at $t = 10$ s, 50 s, 200 s and 500 s. Location of $B_z = 0$ is ~ 2.6 m and it does not change during NBI injection phase.

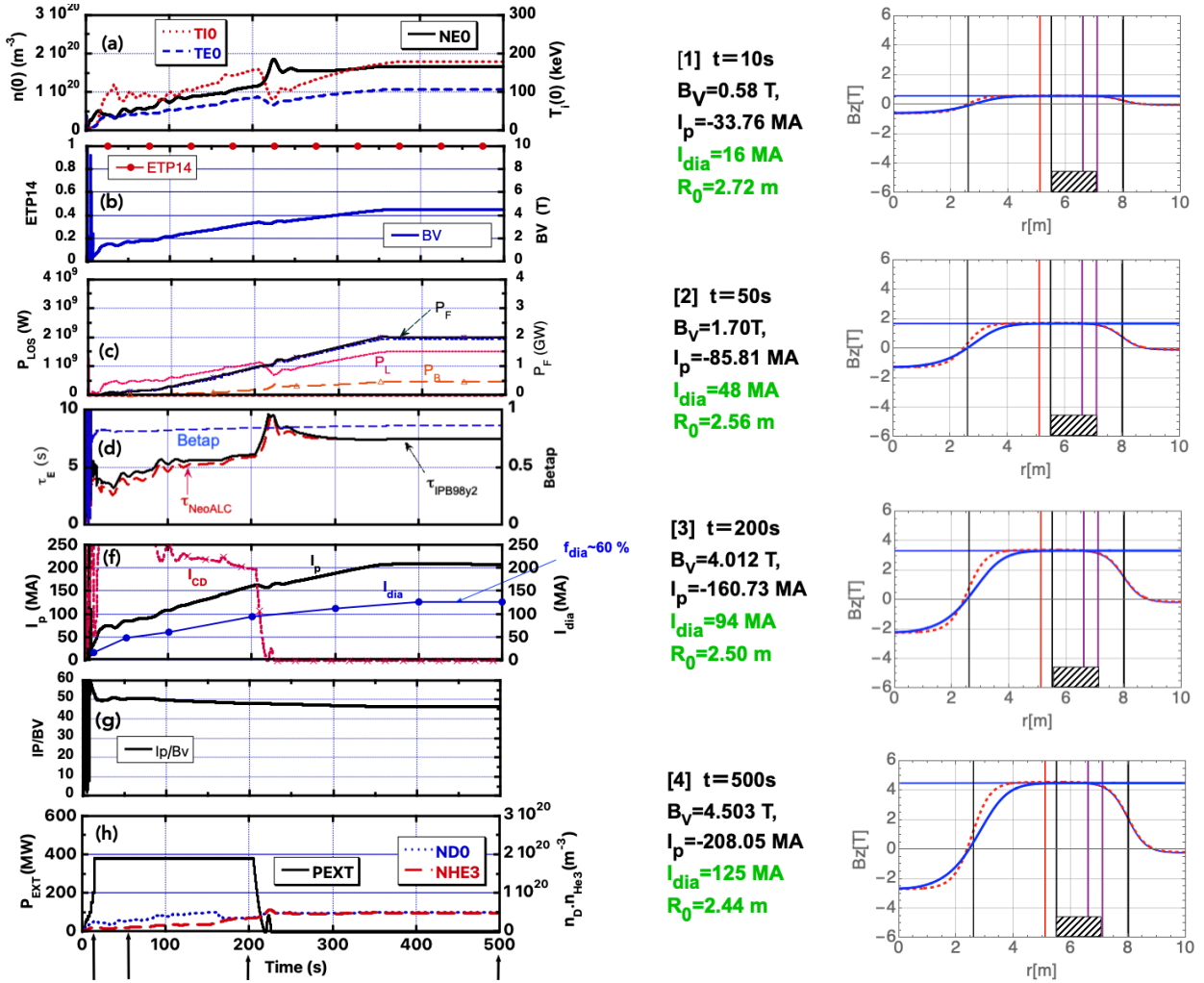


Fig.8 (Left column) Temporal evolution of the plasma parameters during ignition access. (a) NEO: electron density, TIO: peak ion temperature, TEO: peak electron temperature, (b) B_v : axial magnetic field, ETP14: 14 MeV proton loss fraction, (c) P_F fusion power, P_B : total bremsstrahlung loss power, P_L : total plasma conduction loss power, (d) τ_{NeoALC} : NeoAlcaror confinement time, τ_{IPB98y2} : IPB98y2 confinement time with enhancement factor, BETAP the beta poloidal, (e) I_{cd} : NBI driven current, I_p the plasma current, I_{dia} : diamagnetic current, (f) I_p/B_v ratio to see the null point change, (g) PEXT: the external heating power, ND0: peak deuterium density, NHE3: peak ^3He density.

(Right column) Time dependence of the externally applied axial magnetic field (vertical field) B_v , and internal magnetic field profile (blue line) made by plasma current for NBI injection point

4.2 Dependencies of diamagnetic current on density profile

If density profile α_n becomes steeper for the same temperature profile α_T , the peak pressure $p(0)$ increases. Thus, we can have a large diamagnetic current for the peaked pressure profile. We show how a diamagnetic current depends on the peak pressure in Fig. 9, where all the cases have ignition. In the case of $\alpha_n = 0.5$ and $\alpha_T = 0.5$ the diamagnetic current fraction is almost half as seen already in Fig. 8. However, steeper density profile with $\alpha_n = 1.4$ and $\alpha_T = 0.2$ provide the diamagnetic current of 210 MA, providing $\sim 100\%$ of the plasma current. For a flatter density profile than $\alpha_n < 1.4$, the diamagnetic current is less than the total plasma current of ~ 210 MA. As it is found that diamagnetic current depends on the pressure profile, the peaked density profile is required to have a steady state operation. We note that as the FRC density profile based on the RRM is steeper than $\alpha_n = 1.4$ parabolic profile as shown in Fig. 3, actual diamagnetic current may be larger than these values.

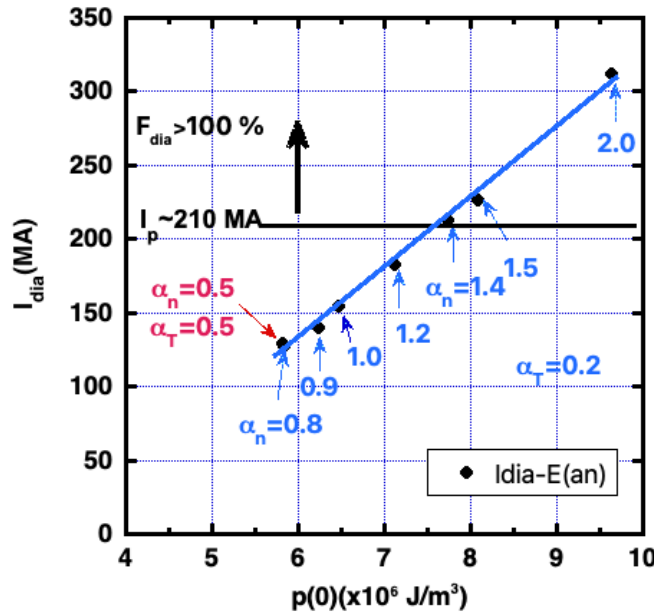


Fig. 9 Diamagnetic current vs the peak pressure $p(0)$ depending on the density profile
The confinement enhancement factor of $\gamma_{HH} = 1.3$.

4.3. Ignition for peaked density profile of $\alpha_n = 1.4$

4.3.1 Complete confinement case of 14 MeV protons

For the peaked density profile of $\alpha_n = 1.4$ and $\alpha_T = 0.2$, ignition access is calculated as shown in Fig. 10 for $R = 2.6$ m, $a = 2.5$ m and $\kappa = 8$. The other setting parameters are the same as in Fig.

8. In general, it is easier for the peaked density profile to access ignition compared to broader density profile [49]. Therefore, confinement enhancement factor is reduced to $\eta_{HH}=1.2$. Steady state parameters are $n(0) = 2.3 \times 10^{20} \text{ m}^{-3}$, $T_i = 156 \text{ keV}$, $T_e = 92 \text{ keV}$, $T_i/T_e = 1.69$, $P_B = 0.46 \text{ GW}$, $P_L = 1.5 \text{ GW}$, $P_n = 36 \text{ MW}$, the plasma current $I_p = 200 \text{ MA}$, the confinement time $\tau_E = 6.8 \text{ s}$ and the ^4He and proton ash fraction $f_{ash} = 3.7 \%$, and the final vertical field $B_v = 4.28 \text{ T}$. The average beta value based on the axial magnetic field is 36.3 %. Diamagnetic current is set to 100 % of the plasma current as obtained in Fig. 9, leading to the steady state operation.

Although we assume the development of the direct energy conversion system in this D- ^3He FRC reactor design, if direct energy conversion is not feasible, we should rely on the thermal conversion alone. In the case of no 14 MeV proton loss, the expected electric power output is $P_E = (0.46+1.5) \times 0.4 = 0.78 \text{ GW}$ by the thermal conversion with efficiency of 40 % for the fusion power of 2 GW. It is assumed that the bremsstrahlung loss power to the first wall is converted to electricity, and the plasma conduction loss power to the divertor is also converted to electricity.

In Fig. 10(g), we show the plasma operating contour map (POPCON) on T_i - n_e diagram from the starting point to the steady state. The steady state is determined by the IPB98y2 confinement time scaling as seen from Fig. 10(d), where the operating point is stopped at the vertical line of the ignition boundary as shown in Fig. 10(g).

The magnetic field profiles between $t = 10 \text{ s}$ and 500 s in Fig. 10 are almost the same as in Fig. 8. This is because the density profile does not affect the magnetic field profile.

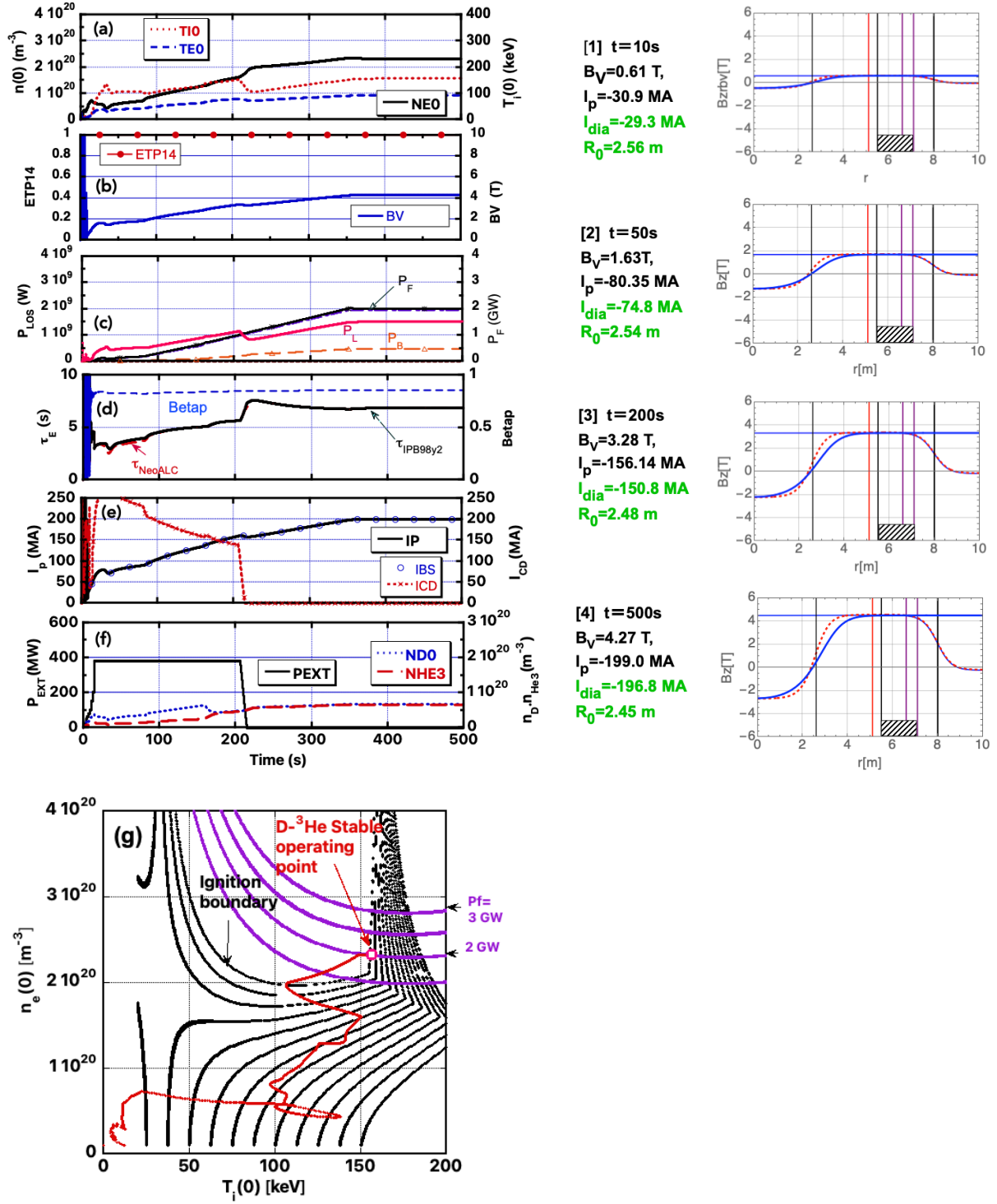


Fig. 10 (a)-(f) Temporal evolution of the plasma parameters during ignition access for $R = 2.6$ m and $a = 2.5$ m. As I_p/B_V ratio is almost the same in Fig. 8, it is not shown here.
 (g) Corresponding POPCON
 (Right column) The magnetic field profiles between $t = 10$ s and 500 s

4.3.2 35 % prompt loss case of 14 MeV proton for direct energy conversion

To increase the electric power output, 14 MeV proton loss leaked from the mirror ends can be

converted to electricity by the traveling wave energy convertor as employed in ARTEMIS [20][21]. Here we assume that the 14 MeV proton end loss is increased from 0 to 35 % of the total 14 MeV proton power ($P_{14\text{MeV}} = 0.73 \text{ GW}$) between $t = 300 \text{ s}$ and 350 s by adjusting the magnetic field at the mirror throat after reaching ignition as shown Fig. 11(b). Other parameters are the same as in Fig. 10 ($\alpha_n = 1.4$, $\alpha_T = 0.2$ and $\gamma_{HH} = 1.2$). Although the 14 MeV proton end loss power increases from 0 to 0.73 GW, the fusion power in the plasma is kept at $P_f = 2 \text{ GW}$ by feedback control. Therefore, actually produced fusion power is increased to 2.73 GW as shown in Fig. 11(c). As fusion power is kept at 2 GW by feedback control, additional 0.73 GW fusion power is immediately lost to the mirror end.

Plasma parameters in the steady state are $n(0) = 3.0 \times 10^{20} \text{ m}^{-3}$, $T_i = 117 \text{ keV}$, $T_e = 83 \text{ keV}$, $T_i/T_e = 1.4$, $\tau_E = 9.2 \text{ s}$ and $f_{\text{ash}} = 5.2 \%$. Various powers are $P_B = 0.73 \text{ GW}$, $P_L = 1.2 \text{ GW}$, and $P_n = 59 \text{ MW}$. Plasma current reaches $I_p = 207 \text{ MA}$, and the vertical field is $B_v = 4.46 \text{ T}$, which are not much different from no 14 MeV proton loss case. As a result of the increase in the density due to the excess fusion power caused by proton loss, the confinement time is increased from $\tau_E = 6.8 \text{ s}$ to 9.2 s , and the bremsstrahlung loss is also increased from $P_B = 0.46 \text{ GW}$ to 0.73 GW . We note that the temporal change of the magnetic field profile at $t = 10 \text{ s}$, 50 s , 200 s and 500 s are almost the same in Fig. 10.

In Fig. 11(g), corresponding POPCON is shown. The operating point moves from the higher temperature regime to the lower temperature regime by increasing 14 MeV proton losses. For the thermal conversion with 40 % and the direct energy conversion with 70 % efficiency [50]-[52], the electric power output is increased as given by $P_E = (0.73+1.2) \times 0.4 + 0.73 \times 0.7 = 1.28 \text{ GW}$.

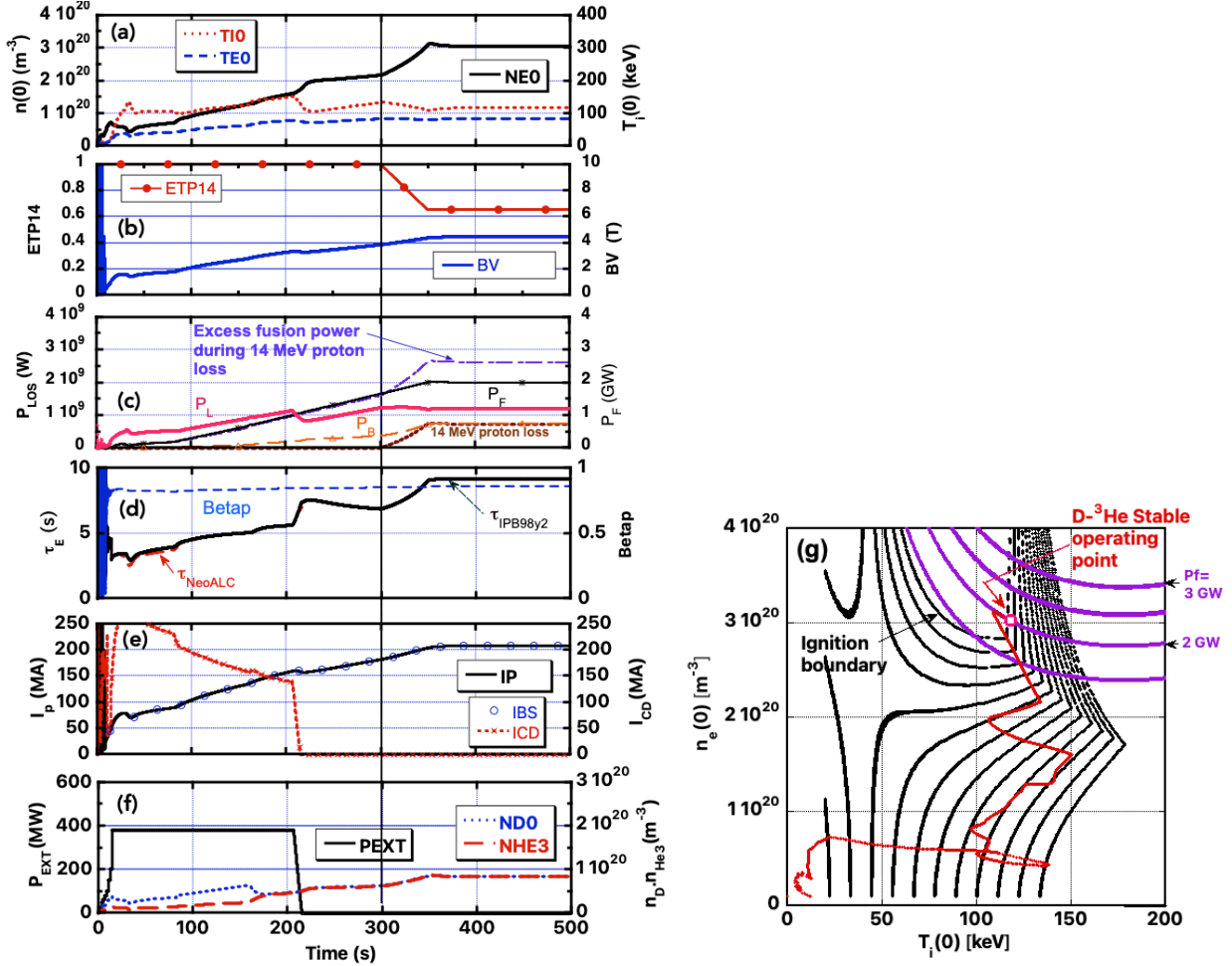


Fig. 11 (a)-(f) Temporal evolution of the plasma parameters during ignition access for $R = 2.6$ m and $a = 2.5$ m. The proton loss increases after 300 s from 0 to 35% linearly to the constant after 350 s. (g) Corresponding POPCON

4.4. Ignition of $R = 3.0$ m and $a = 2.1$ m size for peaked density profile of $\alpha_n = 1.4$

The other configuration with the larger plasma radius but the same total radius ($R + a$) can be considered as $R = 3.0$ m and $a = 2.1$ m. Elongation should be increased to $\kappa = 9.8$ to have the same FRC plasma length and volume. The confinement enhancement factor is reduced to $\gamma_{HH} = 1.0$ because the IPB98y2 scaling has the large major radius dependence. ⁴He and proton ash confinement time ratio is assumed to be $\tau_{ash}/\tau_E = 1$ as before. In Fig.12 are shown the plasma parameter evolutions. The NBI heating and current driver power is now reduced to 300 MW, because the plasma induction effect is enhanced by increase in the vertical field flux due to larger major radius.

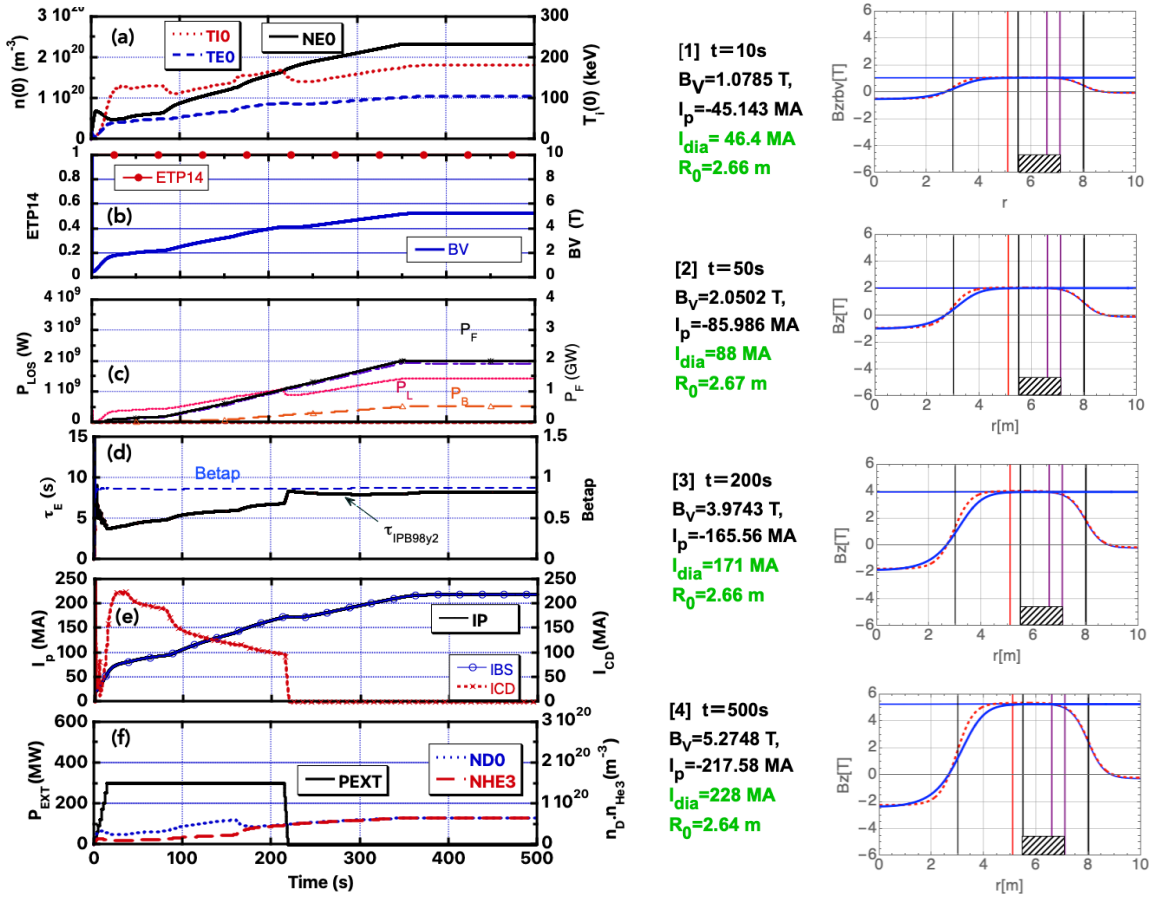
Steady state parameters are $n(0) = 2.32 \times 10^{20}$ m⁻³, $T_i = 181.6$ keV, $T_e = 106$ keV, $T_i/T_e = 1.73$, and $P_B = 0.53$ GW, $P_L = 1.43$ GW, $P_n = 39.5$ MW, the plasma current is $I_p = 217$ MA, confinement time is $\tau_E = 8.2$ s, and ⁴He and proton ash fraction is $f_{ash} = 4.5$ %, and the final vertical field is $B_v = 5.27$ T.

The average beta value based on the axial magnetic field is reduced to 27.7 % because the axial magnetic field is slightly larger. Diamagnetic current is set to 100% of the plasma current, leading to the steady state operation. As this operation does not assume 14 MeV proton loss for energy conversion, the expected electric power output by the thermal conversion with efficiency of 40 % is $P_E = (0.53+1.43) \times 0.4 = 0.785$ GW for the fusion power of 2 GW.

Right column in Fig. 12 shows the temporal change of the magnetic field profile given by B_v and plasma current I_p at $t = 10$ s, 50 s, 200 s and 500 s. Location of $B_z = 0$ is ~ 2.66 m and does not change during NBI injection phase. As the plasma major radius R is increased to $R = 3$ m, the null point is slightly increased than 2.6 m case. Here, the small change in the plasma position needs the small correction factor 1.07 instead of 1.12 in Eq. (11) for adjusting the magnetic field profile

$$B_z(r) = B_o + 1.07 \times 0.5 \times C \times I_p [A] \left\{ \tanh \left(\frac{r^2 - r_o^2}{r_o \Delta r} \right) - 1 \right\} \quad (12')$$

In Fig. 12(g), we show the plasma operating contour map (POPCON) on T_i - n_e diagram from the starting point to the steady state. The ignition region is wider than that in Fig. 10.



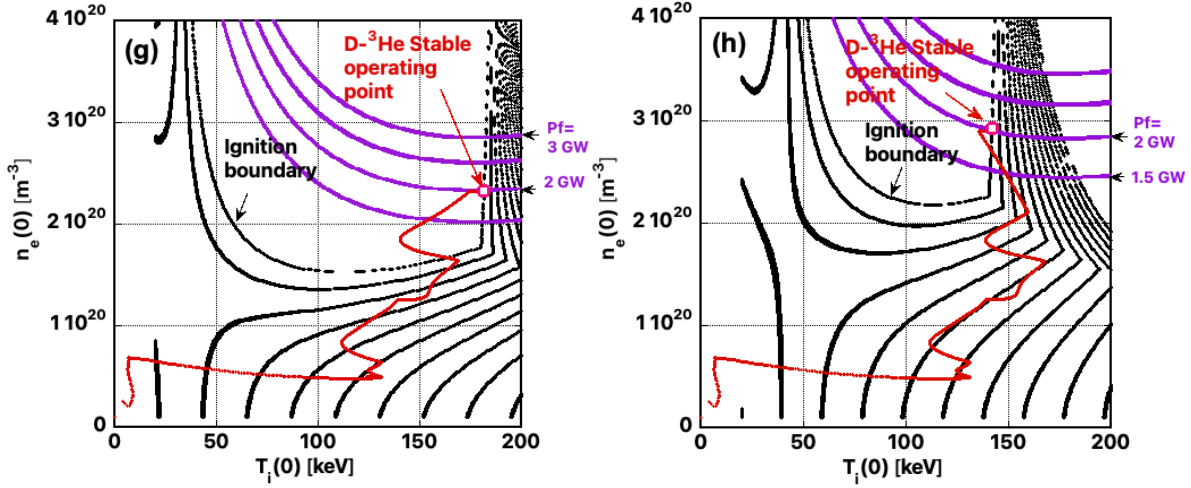


Fig. 12 (a)-(f) Temporal evolution of the plasma parameters during ignition access for $R = 3.0$ m and $a = 2.1$ m.

(g) Corresponding POPCON, and (h) POPCON when 14 MeV proton is lost by 35 %
(Right column) Magnetic field profiles between $t = 10$ s and 500 s.

When 14 MeV proton is lost by 35 % ($P_{14\text{MeV}} = 0.73$ GW) after $t = 300$ s in this configuration, the ignition plasma parameters are changed as listed in Table 1, together with already presented results. As shown in Fig. 12(h), the operating point moves from the higher temperature to the lower temperature regime by reducing 14 MeV proton confinement.

Table 1. Ignition plasma parameters at 500 s in the proposed D-³He FRC reactor.

Parameter at ignition state at 500 s	No 14 MeV proton loss	14 MeV proton loss	No 14 MeV proton loss	14 MeV proton loss
	Fig. 10	Fig. 11	Fig. 12	
Major radius R_o (m)	2.6		3.0	
Minor radius $a = \Delta r$ (m)	2.5		2.1	
Plasma length L (m)	40		40	
Elongation based on ST (κ) and FRC (E)	8.0 (4)		9.8 (4)	
Plasma volume V_o (m ³)	2566		2559	
Confinement enhancement factor η_{HH}	1.2		1.0	
⁴ He and proton ash confinement time ratio	1.0		1.0	
Density profile α_n	1.4		1.4	
Temperature profile α_T	0.2		0.2	
Plasma current I_p (MA)	200	207	217	223.6
Magnetic field B_v (T)	4.28	4.45	5.27	5.42
Diamagnetic current fraction f_{dia} (%)	100	100	100	100
Peak electron density $n(0)$ (m ⁻³)	2.3×10^{20}	3.0×10^{20}	2.32×10^{20}	2.9×10^{20}
Peak ion temperature $T_i(0)$ (keV)	156	117	182	142
Peak electron temperature $T_e(0)$ (keV)	92	83	106	96
Average beta value $\langle \beta \rangle$ %	36.3	36.2	27.7	27.7
Confinement time τ_E (s)	6.8	9.2	8.2	10.5
⁴ He and proton ash fraction f_{ash} (%)	3.7	5.2	4.5	6.3
Fusion power for control P_f (GW)	2.0	2.0	2.0	2.00
Instantaneous fusion power P_{Total} (GW)	2.0	2.73	2.0	2.73

14 MeV proton loss fraction after 300 s	0	35 %	0	35 %
14 MeV proton loss power $P_{14\text{MeV}}$ (GW)	0	0.73	0	0.73
Bremsstrahlung power P_b (GW)	0.46	0.73	0.53	0.76
Synchrotron loss power P_s	0	0	0	0
Plasma conduction loss P_L (GW)	1.5	1.2	1.43	1.18
Neutron power P_n (MW)	36	59	39.5	59
Electric power P_e (GW)	0.78	1.28	0.785	1.28

4.5 Ignition access with low NBI current drive efficiency

As shown in Fig. 5, orbit of NBI ions depend on where they are ionized. Many ions go around to the ion diamagnetic direction, increasing the NBI current drive efficiency. However, other ions go around to the opposite direction, reducing NBI current drive efficiency. Given NBI driven current in this paper so far is very ideal and is maximum. Therefore, we consider how the lower current drive efficiency affects the discharge evolution.

In Fig. 13, the case is shown where the NBI driven current is 1/10 of the ideal case. This calculation parameter is the same as in Fig. 12 with $R = 3.0$ m, $a = 2.1$ m and $\kappa = 9.8$. We do see that the discharge evolutions are totally the same as in Fig. 12 except for the small NBI driven current ~ 10 MA as in Fig. 13(e). This is because the driven current is due to the induction effect of the vertical field. Heating power is most important rather than the driven current in this operation scenario. If the NBI driven current is much lower as $I_{\text{NBI}}/100$, discharge behavior is still the same. Therefore, we do not care about NBI current drive efficiency, but we should care about its heating efficiency.

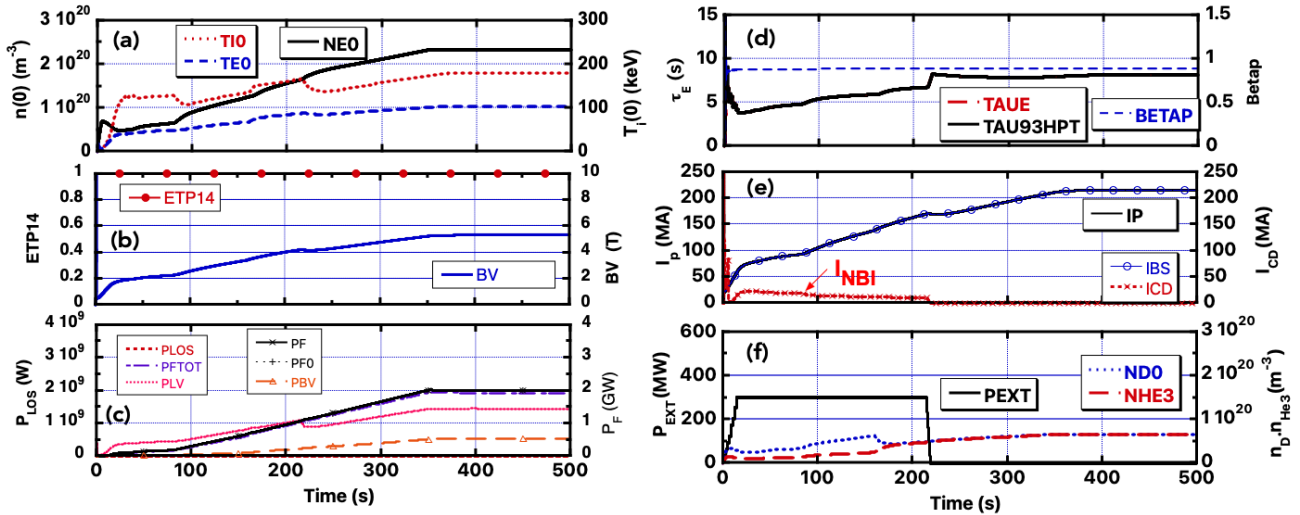


Fig.13 Temporal evolution of the plasma parameters during ignition access for $R = 3.0$ m and $a = 2.1$ m except for the NBI driven current of 1/10.

4.6 Shutdown scenario for the case of full diamagnetic current

When the diamagnetic current is 100 % of the plasma current, the plasma current does not decay, yielding the steady state. Although it seems to be difficult to shut down the discharge, the

plasma current can be decreased by reducing the set value of the the fusion power as shown in Fig. 14. Here the fusion power set value is reduced linearly from 400 s to 480 s (Fig. 14(c)). The discharge behavior until 400 s is the same as Fig. 10. After 400 s the plasma current, density and temperature are all decreased according to the fusion power. All these parameters slowly decay with a delay time of the confinement time. Although the calculated minimum plasma current is ~ 50 MA in the shutdown phase, smoother shutdown can be achieved by adjusting the gas puffing, heating power, and confinement time control, where detailed studies might be necessary. It is also necessary to ensure that the FRC solenoid circuit is equipped with the high voltage protection circuit in case of the disruptive plasma current termination.

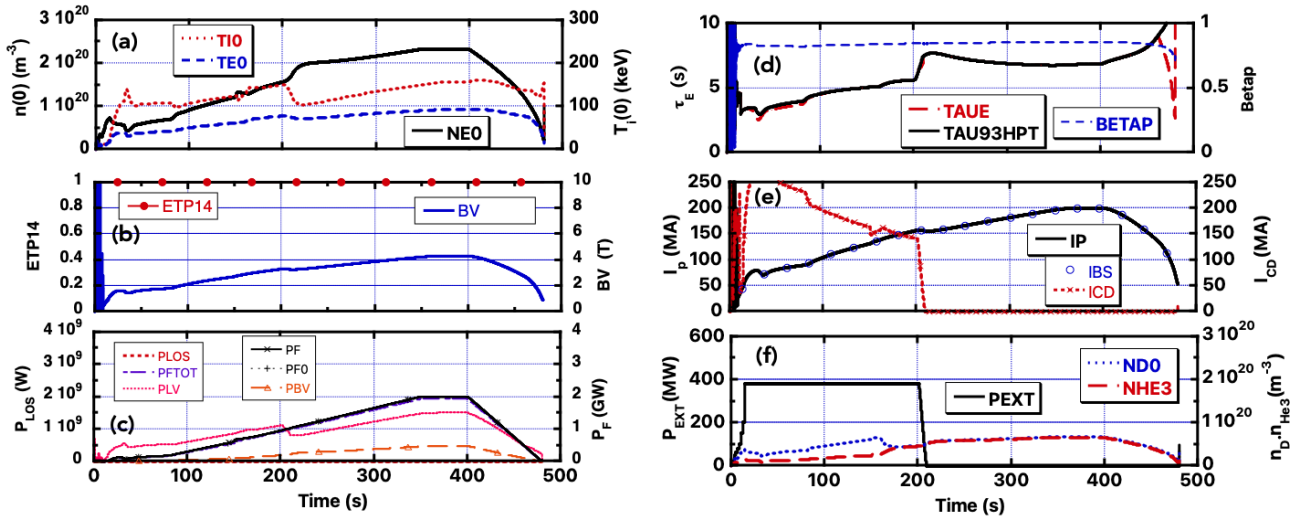


Fig.14 Temporal evolution of the plasma parameters during the shutdown phase after 400 s for $R = 2.6$ m and $a = 2.5$ m. Fusion power set value is 0 at $t = 480$ s.

5. Stability parameters during ignition access

Tilting instability remains a key challenge for sustaining FRC plasmas. In this section, we quantitatively evaluate several stability parameters that have been proposed in previous studies to assess whether ignition conditions in a D-³He FRC reactor meet or exceed current experimental and theoretical stability thresholds. The role of high-energy ions, such as 1.5 MeV NBI deuterons and 14 MeV fusion protons, is also considered. Most critical problem in FRC study at present is the tilting instability. However, optimistic expectation of stability by the high energy particles is described in Reference [53]. It says in page 2071 that "*Although quantitative numbers have not been determined, gross FRC stability might be achieved by maintaining some high energy tail on the ion population. This could occur naturally once ignition is reached, with large-orbit charged fusion products.*" We are encouraged by this statement to estimate the stability parameters in detail including the effect of 1.5 MeV NBI ions and 14 MeV fusion product ions [54][55]. Here we consider three stability parameters described in Reference [56] to view how extrapolation is needed for stability in this D-

³He FRC reactor.

[1] Kinetic $\langle s \rangle$ parameter [57]

The stability parameter $\langle s \rangle$ indicating the average minor radius/ high energy local ion Larmor radius is given by integration between the null point (the major radius) $R_o = 2.44$ and the plasma edge radius (the separatrix radius) $R_s = 2.6+2.5 = 5.1$ m

$$\langle s \rangle = \left\langle \frac{R_s}{\rho} \right\rangle = \int_{R_o}^{R_s} \frac{r dr}{R_s \rho} = \frac{0.695 \times 10^4}{R_s} \frac{Z}{\sqrt{A_{ion} W_{ion}}} \int B_z(r) r dr \quad (15)$$

where $B_z(r)$ is magnetic field profile calculated by Eq. (11) or (11'), the ion Larmor radius ρ is

$$\rho = \frac{V_{\perp ion}}{\Omega_{ion}} \quad \text{with} \quad V_{\perp ion} = 1.38 \times 10^4 \sqrt{\frac{W_{ion}}{A_{ion}}} \quad \Omega_{ion} = 0.96 \times 10^8 Z \frac{B_z(r)}{A_{ion}}$$

$\langle s \rangle$ is calculated for (1) NBI D ion with the energy of $W_{ion} = 1.5$ MeV, the ion mass number $A_{ion} = 2$ and the charge number $Z = 1$, and (2) fusion product proton with $W_{ion} = 14.7$ MeV, $A_{ion} = 1$ and $Z = 1$.

[2] $S = R_o / \rho_o$ [58]

The stability parameter S indicating the major radius R_o / the ion Larmor radius based on the external magnetic field B_e is given by

$$S = R_o / \rho_o = R_o \frac{0.695 \times 10^4}{\sqrt{A_{ion} W_{ion}}} \frac{Z B_e}{\sqrt{A_{ion} W_{ion}}} \quad (16)$$

[3] $S^* = R_s / (c / \omega_{pi})$ [56]

The stability parameter S^* indicating the separatrix radius R_s / the ion inertial length (ion skin depth) is given by

$$S^* = \frac{R_s}{c / \omega_{pi}} = 4.39 \times 10^{-9} R_s \sqrt{\frac{n_i}{A_{ion}}}$$

where ω_{pi} is the ion plasma frequency, c is the speed of light, n_i is the ion density, and

$$c / \omega_{pi} = c / \sqrt{\frac{e^2 n_i}{\epsilon_o M_i}} = \sqrt{\frac{M_i}{\mu_o e^2 n_i}}$$

with e being the elementary charge, ϵ_o being permittivity, μ_o being magnetic permeability and M_i being ion mass.

To calculate this parameter we consider two cases as follows with (1) the mixed fuel ions of D and ³He, and (2) the fusion product 14 MeV proton.

[3.1] Mixed fuel ions

For the mixed fuel ion with D density n_D and ³He density n_{3He} , the equivalent ion mass number is

$$A_{ion} = \frac{2n_D + 3n_{He3}}{n_D + n_{He3}} = 2 \left(\frac{n_D}{n_D + n_{He3}} \right) + 3 \left(\frac{n_{He3}}{n_D + n_{He3}} \right)$$

then

$$S^* = 4.39 \times 10^{-9} R_s \sqrt{\frac{(n_D(0) + n_{He3}(0)) / (1 + \alpha_n)}{2 \left(\frac{n_D}{n_D + n_{He3}} \right) + 3 \left(\frac{n_{He3}}{n_D + n_{He3}} \right)}} \quad (17)$$

where the peaked density is converted to the volume averaged density, and calculated $n_D(0)$ and $n_{He3}(0)$ are used during ignition access.

[3.2] Fusion product proton

For the fusion product 14 MeV proton,

$$S^* = 4.39 \times 10^{-9} R_s \sqrt{\frac{n_p(0) / (1 + 2\alpha_n)}{1}} \quad (18)$$

As the number of the fusion product is proportional to the $n_D n_{He3}$, then we assume the volume averaged density as shown above. Fusion proton density $n_p(0)$ was estimated during the ignition access phase. For example, the ratio of proton to the electron density is $n_p(0)/n_e(0) \sim 1.9\%$ at 500 s.

In Fig. 15(a) is shown the temporal evolution of the stability parameter $\langle s \rangle$ during ignition access for $R = 2.6$ m, $a = 2.5$ m and no 14 MeV proton loss case corresponding to Fig. 10. In Fig. 15(a), the kinetic $\langle s \rangle$ parameter for 14 MeV proton gradually increases and reaches at $\langle s \rangle \sim 12$ at 500 s. On the other hand, the kinetic $\langle s \rangle$ parameter for 1.5 MeV NBI ion increases up to $\langle s \rangle \sim 20$ during NBI injection phase. Experimentally obtained kinetic $\langle s \rangle$ parameter so far in LSX is shown by the shaded region with $\langle s \rangle < 8$ [59], and three-dimensional hybrid simulation provides $\langle s \rangle \sim 7.4$ [60]. These results show the necessity of more experimental and theoretical efforts to demonstrate the stability in the D-³He FRC fusion reactor.

In Fig. 15(a), S parameter for 14 MeV proton provides $S \sim 20$ at 500 s, and S parameter for 1.5 MeV NBI ion reaches more than $S \sim 30$ at 200 s. These values are corresponding $S^{-1} \sim 0.05$ and 0.033, which are in the finite growth rate region of the tilting instability as shown in Fig. 8 in Reference [58].

In Fig. 15(b), S^* parameter for D-³He mixed ions finally reaches $S^* \sim 100$ at 500 s. Similar value is presented in Fig. 5 in Reference [61]. In this FRC ignition study, we assume the elongation of $E = 4$ to take the tilting stability into accounts. In recent FRC experiments such as C-2W [62] and theoretical calculations [63]-[65], the stability criterion of $S^*/E = 3.5 \sim 4$ is given. If we consider this case of $S^* = 100$ as shown in Fig. 15(b), it provides the much larger value of $S^*/E =$

25, which is far from the stability. On the other hand, S^* parameter for 14 MeV proton reaches $S^* \sim 22$ at 500 s. If we employ this high energy particle's value, $S^*/E \sim 5.5$ is obtained which is closer to the presently achieved value of 3.5~4.

Thus, it is not clear at present which stability criterion is best for considering a D^3He FRC reactor.

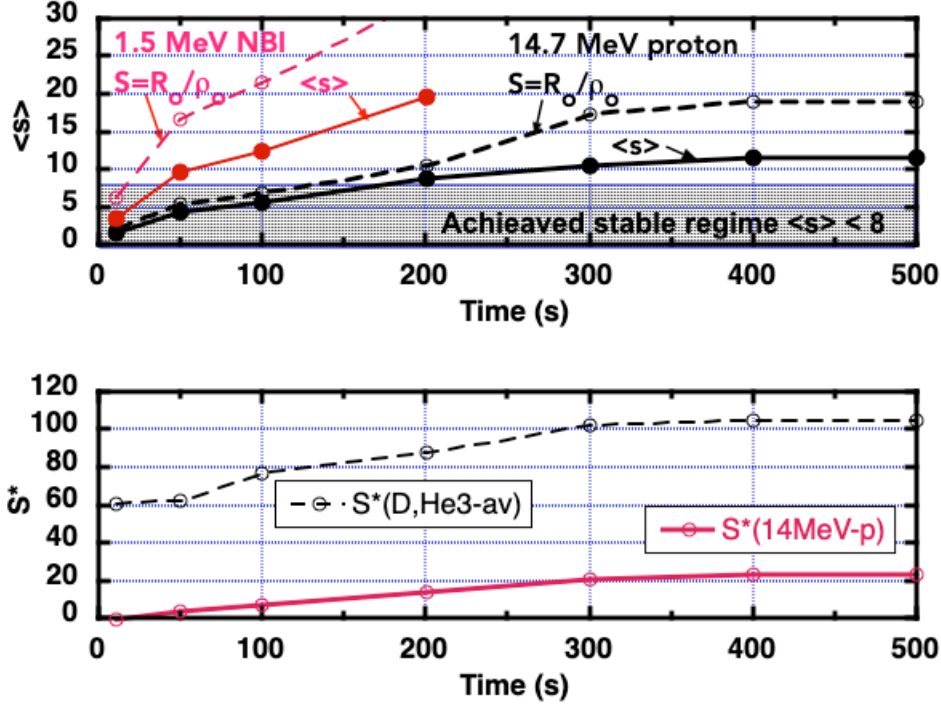


Fig. 15 Stability parameters during discharge for FRC reactor in Fig. 10, where $R = 2.6$ m, $a = 2.5$ m, $I_p = 200$ MA, $B_v = 4.28$ T, the peaked density profile $\alpha_n = 1.4$ and $\alpha_T = 0.2$. (a) $\langle s \rangle$ and S parameters for 14 MeV proton and 1.5 MeV NBI ion, (b) S^* for fuel ions and S^* for 14 MeV proton.

Next we show the temporal evolution of the stability parameters when 14 MeV proton is lost for energy conversion in Fig. 16. After 300 s, $\langle s \rangle$ and S are both increased due to increase in the plasma density for 14 MeV proton losses. Actual fusion power is increased for supplying the excess 14 MeV proton for energy conversion in addition to 2 GW fusion power in the plasma. 14MeV proton density is therefore increased for energy conversion. Thus, all the stability parameters are slightly increased at 400 s as shown in Fig. 17(a) and (b).

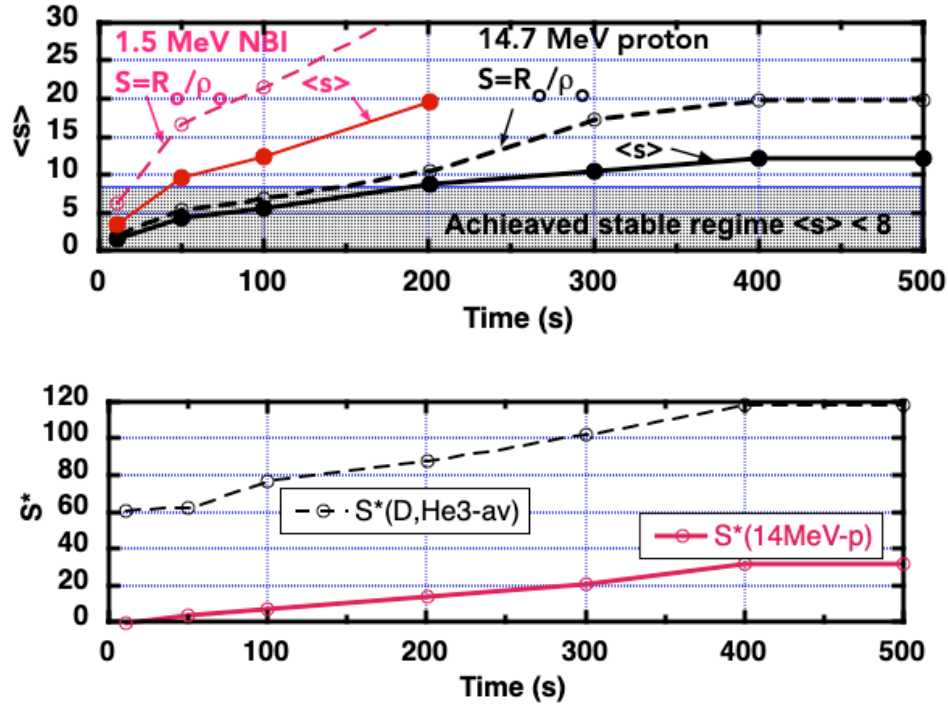


Fig. 16 Stability parameters during discharge in Fig. 11, where 14.7 MeV protons are lost after 300s . $R = 2.6$ m, $a = 2.5$ m, $I_p = 207$ MA, and $B_v = 4.46$ T

If the major radius is slightly increases as $R = 3.0$ m and $a = 2.1$ m, the final vertical field is increased. Then, $\langle s \rangle$ parameter for 1.5 MeV NBI and 14 MeV proton is increased as seen from Eq. (15). Other stability parameters are almost the same as shown in Table 2.

Table 2. Various stability parameters at $t = 200$ s and 500 s, corresponding to table 1

Parameter at ignition state at 500 s	No 14 MeV proton loss	14 MeV proton loss	No 14 MeV proton loss	14 MeV proton loss
Major radius R_o	2.6 m		3.0 m	
Minor radius $a=\Delta r$	2.5 m		2.1 m	
Stability parameters				
$\langle s \rangle$ for 14 MeV proton at $t = 500$ s	11.6	12.1	12.8	13.2
$\langle s \rangle$ for 1.5 MeV NBI ion at $t = 200$ s	19.6	19.6	21.4	21.4
S for 14 MeV proton at $t = 500$ s	19.0	19.8	25.2	26.0
S for 1.5 MeV NBI ion at $t = 200$ s	32.8	32.8	42.7	42.7
S^* for D- ³ He mixed ions at $t = 500$ s	104.8	118.2	104.2	114.1
S^* for 14 MeV proton at $t = 500$ s	23.6	31.8	26.0	34.3

We should note that as all the required stability parameters are beyond the presently obtained value, further experimental and theoretical efforts are needed for achieving a D-³He FRC reactor. While it remains uncertain which stability parameter provides the most appropriate criterion for D-³He FRC reactors, the analyses presented here indicate that the inclusion of high-energy ions plays a potentially significant role in stability enhancement. Further dedicated experiments and numerical

modeling will be essential to validate these trends and guide reactor design.

6. Discussion on further issue

While D-³He tokamak reactors face significant challenges such as disruptions and high plasma currents, FRC-based concepts may offer advantages in terms of operational simplicity and disruption resilience. However, the feasibility of ignition and steady-state operation in a D-³He FRC reactor remains uncertain. This section discusses key technical issues and open questions that must be addressed to realize such a reactor, including current diagnostics, confinement physics, fueling, and system integration.

Tokamak research so far conducted is very useful to consider a FRC reactor. As mentioned in introduction, a high aspect ratio D-³He tokamak reactor has ~50 MA plasma current, and D-³He ST reactor has ~100 MA plasma current. As a D-³He FRC reactor, which is the ultimate ST, has the plasma current of ~200 MA, plasma performance could be better than tokamak. While the plasma current plays an important role for higher performance in a tokamak system, the plasma current seems to be a hidden parameter in a FRC because of no direct measurement. We propose its importance in FRC studies.

6.1 Plasma current measurement

In this study the plasma current is an important factor, but it cannot be directly measured by Rogowski coil. Instead, the flux measurement could be used for estimation. As B_z profile given by Eq. (11) has a I_p dependency, then the measurement of the flux loop (placed at $R_{loop} = 6$ m in this study) encircling the FRC plasma

$$\Delta\Psi_z = \int_0^{R_{loop}} 2\pi r B_z(r) dr \quad (19)$$

could help to estimate the plasma current. The total flux at R_{loop} without the plasma current is

$$\Psi_e = \pi R_{loop}^2 B_e \quad (20)$$

where B_e is the axial (vertical) magnetic field by solenoid. The flux depending on the plasma current alone is given by

$$\Delta\Psi_p = \int_0^{R_{loop}} 2\pi r (B_e - B_z(r)) dr = \Psi_e - \Delta\Psi_z \quad (21)$$

Definition of integration area for these fluxes are shown in Fig. 17.

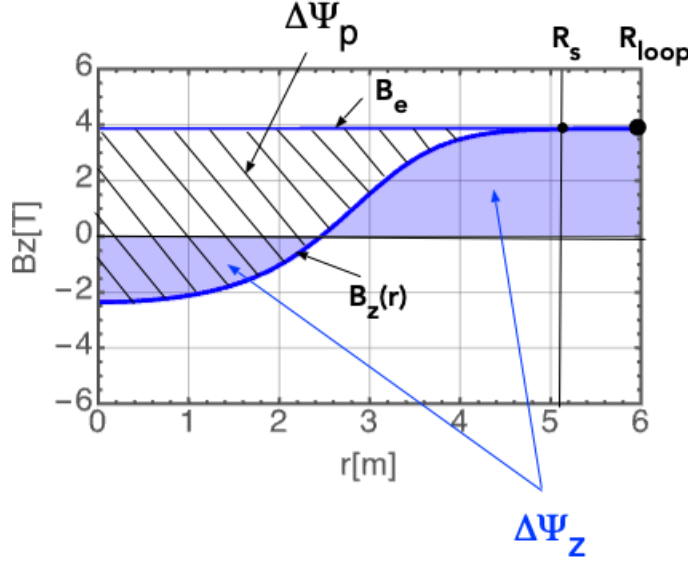


Fig. 17 Various flux definitions (Ψ_e , $\Delta\Psi_z$, $\Delta\Psi_p$) on the magnetic field profile. Flux loop is placed at $R_{\text{loop}} = 6$ m.

The temporal variation of the plasma current and three fluxes (Ψ_e , $\Delta\Psi_z$, $\Delta\Psi_p$) are shown in Fig. 18, where data are taken from Fig. 10. It seems to be possible to estimate the plasma current by the flux measurement. On the other hand, the inductively driven plasma current at 500 s is estimated by the flux relationship of the plasma circuit equation (Eq. (6)) as $I_p = \Psi_p / L_p = 185$ MA where the inductive flux of the plasma current $\Psi_p = (\pi r_o^2) B_e = \pi * 2.6^2 * 4.277 = 84.9$ Vs for the plasma self-inductance $L_p = 0.49 \times 10^{-6}$ H and the small plasma resistance due to D-³He fusion high temperature. Therefore, when we use these flux measurement Ψ_e , the plasma current can be estimated as $I_p = (\pi r_o^2) B_e / L_p = (r_o^2 / R_{\text{loop}}^2) \Psi_e / L_p \sim 0.188 \Psi_e / L_p \sim 0.188 * 485 / 0.49 \times 10^{-6} \sim 185$ MA for $r_o = 2.6$ m and $R_{\text{loop}} = 6.0$ m.

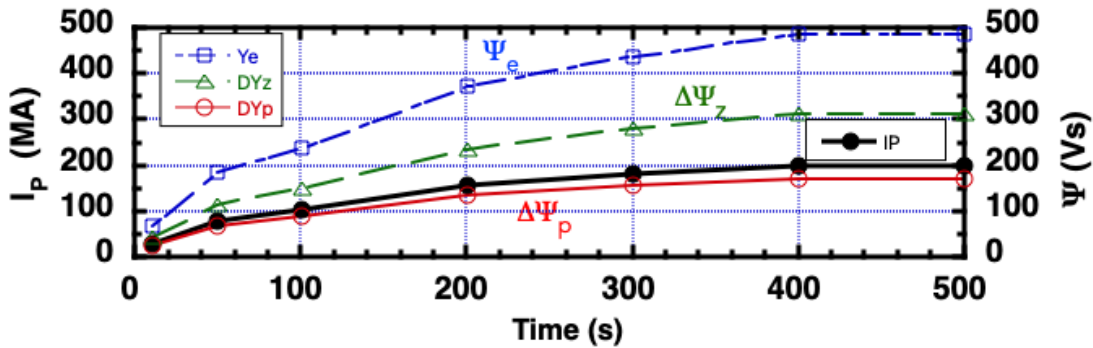


Fig. 18 The time variation of the plasma current and three fluxes (Ψ_e , $\Delta\Psi_z$, $\Delta\Psi_p$) when the flux loop is placed at $R_{\text{loop}} = 6$ m.

If the plasma current is artificially increased from 190 MA to $I_p = 300$ MA at $t = 300$ s without changing the vertical field, $\Delta\Psi_p$ (red line) is increased but $\Delta\Psi_z$ is decreased as shown in Fig. 19. The plasma current alone has been changed in this case. If the vertical field is changed

proportionally at 400 s for equilibrium for the same plasma current of 300 MA, $\Delta\psi_p$ (red line) and $\Delta\psi_z$ (green line) are both increased as shown in Fig. 19. This gedankenexperiments shows that $\Delta\psi_p$ signal responds to the change in the plasma current correctly, but $\Delta\psi_z$ does not respond correctly. If the plasma current density could be separately measured by polarimeter etc, the plasma current can be calibrated by the flux measurement.

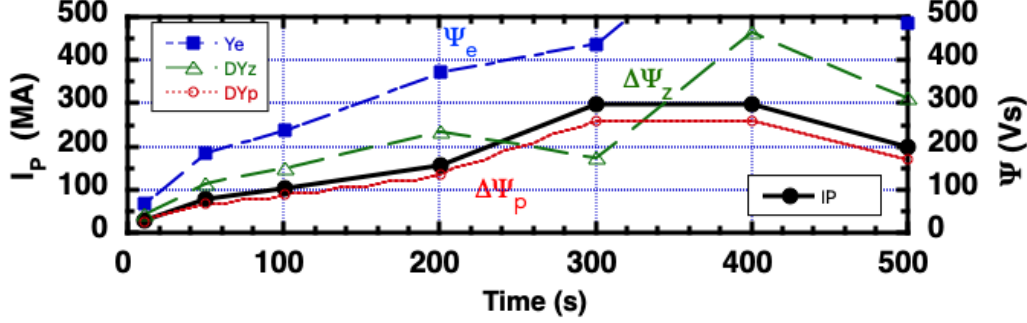


Fig. 19 The time variation of the plasma current and three fluxes (ψ_e , $\Delta\psi_z$, $\Delta\psi_p$). I_p is changed to 300 MA without changing the vertical field at $t = 300$ s, and I_p is changed to 300 MA with changing the vertical field for equilibrium at $t = 400$ s

6.2 Confinement time scaling

On the particle and energy confinement time, the confinement time scaling for FRC has not been established as like tokamak studies. We believe this situation delays ignition studies on FRC although many advantageous points can be found in a FRC fusion reactor. Even at present this situation has not been improved from early 1990s period when Prof. Momota tried to design and calculate D-³He ignition [66] based on the FRC confinement time scaling [67] attached with the temperature term

$$\tau_N(s) = 3.0 \times 10^{-5} \left(r_s(m) / \sqrt{\rho_{io}(m)} \right)^{2.7} T(keV)$$

As this scaling provides a long confinement time as ~ 100 s for this proposed D-³He fusion reactor, we do not use this formula.

On the other hand, IPB98y2 confinement time scaling used for tokamak and spherical tokamak studies is employed in this study because we consider that FRC can be an ultimate extension of a spherical tokamak without the toroidal field. But as no particle drift in FRC due to no toroidal field, confinement physics could be different from tokamak. In this context, we should be careful about this study. However, this study is aiming at knowing what we should do after a FRC can reach D-³He ignition regime. For this purpose, we need the proper confinement time scaling to reach ignition. Fortunately, the IPB98y2 confinement time scaling (Eq. (1)) provides the reasonable value for ignition as you found in this study although its scaling is overpredicting the

confinement data based on NSTX [68].

The FRC the confinement time scaling proposed for compressed FRC fusion reactor is given by [15]

$$\tau_N = 3.2 \times 10^{-15} \varepsilon^{0.5} x_s^{0.8} r_s^{2.1} n^{0.6} \quad (22)$$

where ε is the FRC elongation, n is the density, and r_s is the plasma radius. For the parameters of $\varepsilon = 4$, $n = 2 \times 10^{20} \text{ m}^{-3}$, $r_s = 5.1 \text{ m}$, $x_s = r_s / r_c = 5.1 / 8 = 0.637$ in this D-³He FRC reactor, we have the confinement time of $\tau_N \sim 0.4 \text{ s}$. However, as the plasma current I_p is not included in this scaling (Eq.(22)), then we add the external magnetic field term as B_e / B_{e0} instead of the plasma current, as discussed in Reference [54]. Moreover, in C-2 experiments the high-performance confinement mode has been observed recently, where 10-fold enhancement (γ_{eh}) over the past FRC confinement scaling based on $\tau_N \propto R_s^2 n^{0.5}$ parameter [9]. Adding those terms to Eq. (22) gives the plausible confinement time scaling

$$\tau_N' = \gamma_{eh} 3.2 \times 10^{-15} \varepsilon^{0.5} x_s^{0.8} r_s^{2.1} n^{0.6} (B_e / B_{e0}) \quad (23)$$

Actually for $\gamma_{eh} = 10$ and $B_e = 4 \text{ T}$ and $B_{e0} = 1 \text{ T}$ in this D-³He FRC reactor, we have the confinement time of $\tau_N' \sim 8.2 \text{ s}$ which is comparable to our results ($\tau_E = 6 \sim 10 \text{ s}$). Although this is only the ad hoc calculation, the positive result encourages our studies.

On the other hand, the electron confinement time proportional to $T_e^{1.6}$ has been observed in C-2 [9]. This is quite interesting if its confinement time is interpreted to be the global confinement time. As pointed out in Reference [9], $\tau_{Ee} \propto T_e(0)^{-1}$ provides the Bohm type confinement time. Here, let's assume that the electron confinement time is the global confinement time. We show that $\tau_E \propto T_e(0)^{-1}$ comes from $\tau_E \propto 1 / \sqrt{P_{heat}}$. Such confinement scaling is Goldston scaling [69], which has the power law $S = 0.5$ of the heating power as given here

$$\tau_E [s]_{Goldston} = \frac{6.4 \times 10^{-8} I_p (A) R(cm)^{1.75} a(cm)^{-0.37} \kappa^{0.5}}{\sqrt{P_{heat} (W)}}$$

where I_p is the plasma current, R is the major radius, a is the minor radius, κ is the elongation and P_{heat} is the heating power. By replacing the heating power with the net heating power into the plasma $P_{heat} = P_{NET} (W / m^3) V_o (m^3)$ (V_o being the plasma volume) and the coefficient is given by

$$\gamma_H C_{op} = \gamma_H 6.4 \times 10^{-8} I_p (A) R(cm)^{1.75} a(cm)^{-0.37} \kappa^{0.5} / \sqrt{V_o (cm^3)}$$

the confinement time for D-T fusion is simply expressed as

$$\tau_E = \frac{\gamma_H C_{op}}{P_{NET}^S} = \frac{\gamma_H C_{op}}{(P_{ht} + P_{\alpha} - P_b)^S} \quad (24)$$

where P_{ht} is the external heating power, P_{α} is alpha heating power, and P_b is the bremsstrahlung

power [49]. The power balance equation in the steady state per unit volume

$$\frac{dW}{dt} = (P_{ht} + P_{\alpha} - P_b) - P_L = P_{NET} - P_L = 0$$

provides that the net heating power is equal to the plasma conduction loss as $P_{NET} = P_L$. Here the plasma conduction loss P_L is defined by the plasma energy W_p divided by the confinement time τ_E

$$P_L = \frac{W_p}{\tau_E} = \frac{A_L n(0) T_e(0)}{\tau_E} \quad (25)$$

where

$$A_L = \frac{1.5(f_{De} + f_{Te} + 1)T_e(0) \times 1.6 \times 10^{-19}}{(\alpha_n + \alpha_T + 1)}$$

with $f_D = T_D(0)/T_e(0)$ and $f_T = T_T(0)/T_e(0)$ are the D and T ion temperature ratio to the electron temperature, respectively. By replacing Eq. (25) into Eq. (24), we have

$$\tau_E = \frac{\gamma_H C_{op}}{P_L^S} = \frac{\gamma_H C_{op}}{\left(\frac{A_L n(0) T_e(0)}{\tau_E} \right)^S} = \frac{\gamma_H C_{op}}{(A_L n(0) T_e(0))^S} \tau_E^S$$

and then we finally obtain the confinement time

$$\tau_E = \frac{(\gamma_H C_{op})^{\frac{1}{1-S}}}{(A_L n(0) T_e(0))^{\frac{S}{1-S}}} = (\gamma_H C_{op})^{\frac{1}{1-S}} (A_L n(0) T_e(0))^{\frac{-S}{1-S}} \propto T_e(0)^{\frac{-S}{1-S}} \quad (26)$$

which provides $\tau_E \propto T_e(0)^{-1}$ for $S = 0.5$.

Now, question arises what is the original scaling corresponding to $\tau_E \propto T_e(0)^{1.6}$. If we use the same technique, Eq. (25)

$$\frac{-S}{1-S} = 1.6$$

provides $S = 2.67$, yielding

$$\tau_E = \frac{\gamma_H C_{op}}{P_{heat}^{2.67}}$$

This shows the rather strong confinement degradation by the heating power, which is contrary to our intuition. Thus, we do see that we cannot use the same procedure.

The function $-S/(1 - S)$ is shown in Fig. 20, where $0 < S < 1$ provides the inverse power law $\tau_E \propto 1/P_{heat}^S$ and $S < 0$ provides the positive power law $\tau_E \propto P_{heat}^S$.

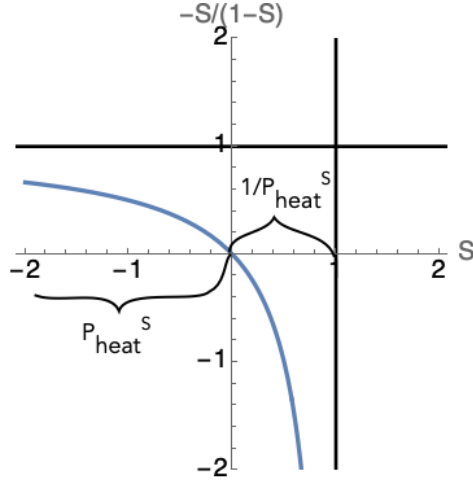


Fig. 20 The function $-S/(1 - S)$ for expressing Eq. (26).

As $-S/(1 - S)$ cannot be over 1.0 as seen from Fig. 20, we have to re-assume the coefficient with $T(0)^{S2}$ term as

$$\tau_E = \frac{\gamma_H C_{op}}{P_{heat}^{S1}} \cdot T(0)^{S2} \quad (27)$$

which provides

$$\tau_E = \left(\gamma_H C_{op} \right)^{\frac{1}{1-S1}} T(0)^{\frac{S2}{1-S1} + \frac{-S1}{1-S1}}$$

For example, if we set $S1=0$,

$$\frac{S2}{1-S1} + \frac{-S1}{1-S1} = S2 = 1.6$$

we obtain

$$\tau_E = \frac{\gamma_H C_{op} \cdot T(0)^{1.6}}{P_L^0} = \left(\gamma_H C_{op} \right) T(0)^{1.6} \quad (28)$$

which has no heating power term. Next if we set $S1= -1.0$,

$$\frac{S2}{1-S1} + \frac{-S1}{1-S1} = \frac{S2}{2} + \frac{1}{2} = 1.1 + 0.5$$

we obtain

$$\tau_E = \frac{\gamma_H C_{op} \cdot T(0)^{2.2}}{P_L^{-1}} = \gamma_H C_{op} T(0)^{2.2} P_L \quad (29)$$

which depends on the heating power and coefficient. There are many other combinations of $S1$ and $S2$. The simplest form is Eq. (28) as demonstrated in Reference [9]. Before considering above result, we would like to confirm that the coefficient C_{op} in $\tau_E = \left(\gamma_H C_{op} \right) T_e(0)^{1.6}$ is omitted when the

experimental data is analyzed. This is because the confinement time $\tau_E = (\gamma_H C_{op}) T_e(0)^{-2.26}$ obtained from the IPB98y2 scaling is increased with $T_e(0)$ due to the coefficient C_{op} . Although $T_e(0)^{-2.26}$ term is decreased with $T_e(0)$, τ_E itself is increased with $T_e(0)$ due to the coefficient C_{op} . This was confirmed by plotting the calculated confinement time and $T_e(0)$ obtained in Fig. 10.

Thus, if this FRC confinement time scaling is established, it is quite interesting because it is entirely different from the tokamak confinement time scaling. Especially, if the heating power and density dependencies are also established, ignition characteristics could be expected.

To establish the confinement time scaling in FRC, multi-machine platform should be created in a world wide as a tokamak community. Recent "Norman" experiments in TAE demonstrates that the initial FRC plasma formation can be done by NBI and side plasma gun injections instead of two side theta pinch coils [70]. This result may show that the mirror machine can be converted to FRC machine with the help of NBI. Therefore, multi-machine environment could be created in a relatively short time. We hope that this ignition study stimulates such research investment and possibility of establishing the multi-machine environments for pursuing FRC fusion reactor.

6.3 Other issues

Diagnostics and fusion control are also big challenges to achieve a D-³He FRC reactor. For example, is it possible to measure the electron temperature among the powerful bremsstrahlung radiation environment? In this study we are using the fusion power as a guide parameter for a control knob. However, while the neutron power can be used to estimate the fusion power in a D-T fusion reactor, it is difficult to estimate the fusion power in a D-³He fusion reactor because main fusion products are the charged particles. Therefore, the different control guide parameters should be considered.

Most difficult task in the D-³He FRC is "fueling" as in a D-³He tokamak reactor. (1). Original idea in ARTEMIS is the "Pacman" method [20]. Cold ice pellet is injected and FRC plasma is moved quickly axially to eat pellets. From the both ends, pellet could be injected almost axially, and FRC plasma moves in the oscillating manner to eat the ice pellets. (2) Injection of CT rings along the axial direction could be a natural method for FRC reactor as demonstrated experimentally [44]-[47]. Repetitive injections of CT rings from the one or both sides is necessary with the particle confinement time ~ 10 s. We pursue this method in detail in future. The last method is (3) the traditional CT injection from the side blanket. However, the CT plasma should be created in the high field area, which might be difficult.

The main concept of this proposed D-³He FRC reactor is taken over from former ARTEMIS

with energy conversion system [20]. In this context, “Neo-ARTEMIS” may be the natural name although its size and operation scenario are much improved. 3-D schematic view of the central core part of Neo-ARTEMIS is shown in Fig. 21. 20 NBI injectors are installed at the oblique angle at the upper and lower parts. Therefore, as 3-D NBI ion orbits should be calculated, but it is beyond this work. Plasma formation, repetitive fueling and energy conversion parts should be combined. We have to consider the possible layout of such combinations.

Finally we hope that this fusion-ARTEMIS could stimulate the NASA Artemis program [71] to extract ^3He resources from the moon regolith, which is an important factor of D- ^3He fusion reactor development.

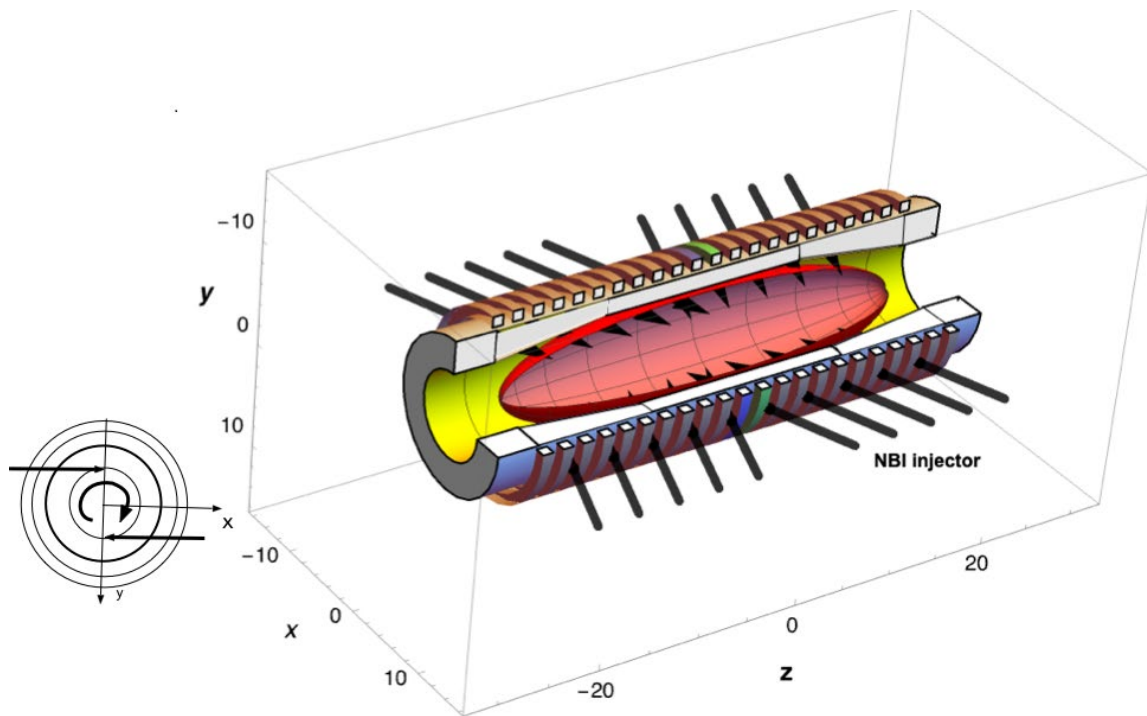


Fig. 21 Schematic view of central part in Neo-ARTEMIS

7. Summary

Based on the assumption that a FRC could be the ultimate ST without the toroidal field and by borrowing the confinement time scaling of ST, ignition access of D- ^3He FRC reactor has been studied to consider what we should do for ignition and further exploration. Although the assumptions--such as applying ST confinement scaling to FRC--are somewhat speculative, the analysis yields valuable insights into the ignition and operation of D- ^3He FRC reactors.

Summaries are shown here.

[1] The proposed D- ^3He FRC reactor is $R = 2.6$ m, $a = 2.5$ m, $E = 4$ and $I_p \sim 200$ MA. With the help of NBI current drive (1.5 MeV, 380 MW) injected at $B_z = 0$ in the initial phase, the plasma current

is mainly ramped up by plasma heating and the vertical field, and finally reached to ~ 200 MA. NBI heating can be switched off at 200 s, leading to ignition.

- [2] As the diamagnetic current for the peaked density profile of $\alpha_n > 1.4$ provides almost 100% of the plasma current, then steady state operation could be possible. However, as a FRC density profile has the more peaked density profile than $\alpha_n = 1.4$ based on Rigid Rotor Model, it could achieve 100% diamagnetic current. Of course, we need detailed studies on density profile effect on ignition, which is our future work.
- [3] Less than 35 % of 14 MeV proton end loss from the mirror end could be converted to electricity under the condition of ignition, owing to the good confinement in this study.
- [4] Plasma current ramp-up without NBI current drive function is also possible to reach ignition in the D-³He FRC with $R = 3$ m, $a = 2.1$ m and $E = 4$. Heating function is important to ramp-up the plasma current.
- [5] During this study it has been found that the experimental and theoretical evidence of the stabilization of the tilting instability is not enough. More experimental and theoretical efforts are needed to guarantee the tilting stability.

Finally, as a natural extension of this work, we propose applying the same framework to investigate D-T FRC ignition, which may clarify the development path from current experiments to a future FRC reactor.

Acknowledgement

We acknowledge Prof. Y. Kishimoto in Kyoto University and Prof. Emeritus T. Takahashi in Nihon University for useful discussions. This work was carried out under the collaborating research program at Nihon University. This work was also carried out under the collaborating research program at National Institute for Fusion Science (NIFS22KIGA002).

References

- [1] O. Mitarai, "Quest for D-³He, D-D advanced-fuel fusion reactors", *J. Plasma Fusion Res.* **98**, (2022) 72 (In Japanese)
- [2] O. Mitarai, H. Matsuura and Y. Tomita, "Aspect ratio dependencies of D-³He fueled tokamak reactors", *Fusion Engineering and Design*, **81** (2006), 2719
- [3] O. Mitarai, H. Matsuura, T. Omori, et al., "Ignition studies of D-³He spherical tokamak reactor" *Fusion Engineering and Design*, **136** (2018) 82
- [4] O. Mitarai, A. Sagara, N. Ohyaibu, et al., "New control method of the unstable operating point in the FFHR helical reactor", *Plasma and Fusion Research, Rapid Communication*, **2** (2007)

- [5] O. Mitarai, A. Sagara, N. Ohyaibu, et al., "Control algorithm of the unstable operating point in the FFHR helical reactor", *Fusion Science and Technology* **56** (2009) 1495
- [6] O. Mitarai, A. Sagara, R. Sakamoto et al., "High density, low temperature ignited operations in FFHR, *Plasma and Fusion Research*, **5** (2010) S1001
- [7] O. Mitarai, S. Sugiyama, N. Yanagi, et al., "Comparative studies on the control algorithm for the high-density ignition regime in FFHR-d1", *Plasma and Fusion Research*, **15**,(2020) 2405059
- [8] O. Mitarai, A. Sagara, S. Imagawa et al., "Parameter requirements for D-³He helical reactors", *Journal of Plasma and Fusion Research*, **6** (2004) 303
- [9] M. W. Binderbauer, T. Tajima, L. C. Steinhauer, et al., "A high performance field-reversed configuration", *Phys. Plasmas* **22**, (2015) 056110
- [10] M. W. Binderbauer, T. Tajima; M. Tuszewski, et al., " A Recent breakthroughs on C-2U: Norman's legacy", *AIP Conference Proceedings* **1721**, (2016) 030003
- [11] H. Gota , M.W. Binderbauer, T. Tajima, et al., "Achievement of field-reversed configuration plasma sustainment via 10 MW neutral-beam injection on the C-2U device", *Nucl. Fusion*, **57** (2017) 116021
- [12] H. Gota , M.W. Binderbauer , T. Tajima, S. et al., "Formation of hot, stable, long-lived field-reversed configuration plasmas on the C-2W device", *Nucl. Fusion*, **59** (2019) 112009
- [13] H. Gota , M.W. Binderbauer , T. Tajima, et. al., "Overview of C-2W: high temperature, steady-state beam-driven field-reversed configuration plasmas", *Nucl. Fusion*, **61** (2021) 106039
- [14] H. Gota, A. Smirnov, M.W. Binderbauer, et al., "Enhanced plasma performance in C-2W advanced beam-driven field-reversed configuration experiments", *Nucl. Fusion*, **64** (2024) 112014
- [15] D. Kirtley, R. Milroy, "Fundamental scaling of adiabatic compression of field reversed configuration Thermonuclear fusion plasmas", *Journal of Fusion Energy* (2023) 42:30
- [16] W. T. Armstrong, R. K. Linford; J. Lipson et al., "Field-reversed experiments (FRX) on compact toroids", *Physic of Fluids*, **24** (1981) 2068.
- [17] O. Mitarai and Y. Takase, "Plasma current ramp-up by the outer vertical field coils in a spherical tokamak reactor", *Fusion Science and Technology*, **43** (2003) 67
- [18] ITER Physics Expert Groups on Confinement and Transport and Confinement Modelling and Database, ITER Physics Basis Editors and ITER EDA, "Chapter 2; Plasma confinement and transport", *Nucl. Fusion*, **39** (1999) 2175
- [19] R. J. Goldston, "Energy confinement scaling in tokamaks: Some implications of recent experiments with ohmic and strong auxiliary heating," *Plasma Phys. Controlled Fusion*, **17**

(1984) 87

- [20] H. Momota, M. Okamoto, and Y. Nomura et al., "Advanced fuels on a field reversed configuration", *Fusion Technology*, **11** (1987) 436
- [21] H. Momota, A. Ishida, Y. Kohzaki, et al., "Conceptual design of the D-³He reactor ARTEMIS", *Fusion Technology*, **21** (1992) 2307
- [22] W. Kernbichler, "Operational parameters for D-³He in field-reversed configurations", *Fusion Technology*, **21** (1992) 2297
- [23] Y. Hirano, J. Sekiguchi, T. Matsumoto, T. Asai, et al., "A D-T fusion reactor design in field-reversed configuration using normal conductive coils", *Nucl. Fusion*, **58** (2018) 0160047
- [24] J. Slough, G. Votroubek and C. Pihl, "Creation of a high-temperature plasma through merging and compression of supersonic field reversed configuration plasmoids", *Nucl. Fusion*, **51** (2011) 053008
- [25] O. Mitarai, "A D-³He spherical tokamak reactor with the plasma current ramp-up by vertical field" in "*Nuclear Reactors, Nuclear Fusion and Fusion Engineering*," Nova Science Publishers, Inc. (2009) 405, Edited by A. Aasen and P. Olsson.
- [26] H. Matsuura, Y. Nakao, K. Kudo and O. Mitarai, "Effect of nuclear elastic scattering on plasma confinement condition in D-³He tokamak fusion energy systems", *12th International Conference on Emerging Nuclear Energy Systems (ICENES'2005)* Brussels, Belgium, August 21–26, 2005
- [27] N. Rostoker and A. Qerushi, "Equilibrium of field reversed configuration with rotation. I. One space dimension and one type of ion", *Physics of Plasmas*, **9**, (2002) 3057
- [28] O. Mitarai, S.W. Wolfe, A. Hirose and H. M. Skarsgard, "Stable AC Tokamak Discharges in the STOR-1M Device," *Nuclear Fusion*, **27** (1987) 604.
- [29] O. Mitarai, A. Hirose and H. M. Skarsgard, "AC Tokamak Reactor with Long Pulses," *Fusion Technology*, **15** (1989), 204
- [30] O. Mitarai "Inductive plasma current start-up by the outer vertical field coil in a spherical tokamak", *Plasma Phys. Control. Fusion*, **41** (1999) 1469
- [31] R. Kurihara, S. Nishio, S. Ueda, et al "Control of fusion power in a steady-state tokamak reactor", *Journal of Plasma and Fusion Research Series*, **3** (2000) 553
- [32] O. Mitarai, R. Yoshino and K. Ushigusa "Plasma current ramp-up assisted by outer vertical field coils in a high aspect ratio tokamak " *Nucl. Fusion*, **42** (2002) 1257
- [33] O. Mitarai, Y. Takase, A. Ejiri, et al., "Plasma current start-up by ECW and vertical field in the TST-2 spherical tokamak", *Journal of Plasma and Fusion Research*, **80** (2004) 549
- [34] M. Inomoto, T.G. Watanabe, K. Gi, et. al., "Centre-solenoid-free merging start-up of spherical tokamak plasmas in UTST", *Nucl. Fusion*, **55** (2015) 033013

- [35] O. Mitarai, Y. Ding, M. Hubeny, et al., "Plasma current sustainment after iron core saturation in the STOR-M tokamak" *Fusion Engineering and Design*, **89** (2014) 2467
- [36] O. Mitarai, C. Xiao, D. McColl, et al., "Plasma current start-up by the outer ohmic coil in the STOR-M iron core tokamak", *Review of Scientific Instruments*, **86** (2015) 033508
- [37] O. Mitarai, G. Tomney, E. Lewis, et al., "Plasma Current Start-up Experiments without Central Solenoid in the Iron Core STOR-M Tokamak", *Nucl. Fusion*, **55** (2015) 063034
- [38] P. Hirshman and G.H. Neilson, "External inductance of an axisymmetric plasma", *Phys. Fluids*, **29**, (1986) 790
- [39] A. F. Lifschitz, R. Farengo and A.L. Hoffman, "Calculations of tangential neutral beam injection current drive efficiency for present moderate flux FRCs", *Nucl. Fusion*, **44** (2004) 1015
- [40] A. L. Hoffman, L.C. Steinhauer, H. Ferrari and R. Farengo, "Advances in singly connected closed field line plasma devices and extrapolation to POP level experiments and reactors", *Nucl. Fusion* **49** (2009) 055018
- [41] J. Wesson, "Tokamaks", Clarendon Press-Oxford (1997).
- [42] S. Nagao, "Plasma equilibrium in self-sustaining toroidal discharge", *Kakuyugou-Kenkyu* **59** (1988) 296
- [43] M. W. Binderbauer, H.Y. Guo, M. Tuszewski, et al., "Dynamic formation of a hot field reversed configuration with improved confinement by supersonic merging of two colliding high- β compact toroids", *Phys. Rev. Lett* **105**, (2010) 045003
- [44] T. Asai¹, D. Kobayashi¹, T. Seki¹ et al., "Observation of self-organized FRC formation in a collisional-merging experiment", *Nucl. Fusion* **61** (2021) 096032
- [45] D. Kobayashi, and T. Asai, "Experimental evidence for super-Alfvenic acceleration of the field-reversed configuration due to a magnetic pressure gradient", *Phys. Plasmas* **28**, (2021) 022101
- [46] T. Asai¹, T. Takahashi, D. Kobayashi et al., "Refueling of field-reversed configuration core via axial plasmoids injection", *Nucl. Fusion* **64** (2024) 096013
- [47] T. Asai, T. Seki, D. Kobayashi, et al., "Observation of rapid flux coalescence in merging field-reversed configurations", *Phys. Plasmas* **31**, (2024) 010703
- [48] O. Mitarai, A Hirose and H M. Skarsgard, "Saddle point condition for D-³He tokamak fusion reactors", *Fusion Technology* **19** (1991) 234
- [49] O. Mitarai, A. Hirose and H. M. Skarsgard, "Operation path method for ignition criterion in a D-T tokamak reactor," *Fusion Technology*, **22** (1992) 227.
- [50] H. Momota, "Direct energy conversion for 15 MeV fusion protons," *US-Japan Workshop on*

Field-Reversed Configurations with Steady-State High-Temperature Fusion Plasmas and the 11th US-Japan Workshop on Compact Toroids (1989) 8, LA-11808-C, Los Alamos National Laboratory, (1990),

- [51] Y. Tornita, L.Y. Shu and H. Momota, "Direct energy conversion system for D-³He fusion", *NIFS-252* Nov. 1993, National Institute for Fusion Science, Nagoya, Japan
- [52] H. Takeno, K. Shibata, S. Nakamoto, et al., "Examination of efficiency dependence on deceleration voltage in a traveling wave direct energy converter simulator", *Plasma and Fusion Research*, **18** (2023) 2405053
- [53] M. Tuszewski, "Field Reversed Configurations," *Nucl. Fusion*, **28** (1988) 2033.
- [54] D. C. Barnes and R.D. Milroy, "Stabilization of the field-reversed configuration (FRC) tilt instability with energetic ion beams", *Phys. Fluids*, **B 3** (1991) 2609
- [55] H. L. Berk, H. Momota, and T. Tajima, "Plasma current sustained by fusion charged particles in a field-reversed configuration", *Phys. Fluids* **30** (1987) 3548. .
- [56] L. C. Steinhauer, "Review of field reversed configurations", *Physics of Plasmas*, **18** (2011) 0070501
- [57] J. T. Slough, A.L. Hoffman, R.D. Milroy, et al., "Flux and particle life-time measurements in field-reversed configurations", *Nucl. Fusion* **24** (1984) 1537
- [58] D. C. Barnes, J.L. Schwarzmeier, H.R. Lewis, "Kinetic tilting stability of field-reversed configurations", *Phys. Fluids* **29** (1986) 2616.
- [59] J. T. Slough, A. L. Hoffma, R. D. Milroy, et al., "Confinement and stability of plasmas in a field-reversed configuration", *Physical Rev. Lett*, **69** (1992) 2212
- [60] E.V.Belova, R.C.Davidson, H.Ji, and M.Yamada, "Kinetic effects on the stability properties of field-reversed configurations, II Nonlinear evolution", *Physics of Plasmas* **11** (2004) 2523
- [61] A. Ishida, R. Kanno, and L.C. Steinhauer, "Tilt stability of a gyroviscous field-reversed configuration with realistic equilibria", *Phys. of Fluids* **B 4**(1992) 1280
- [62] H.Y. Guo, M.W. Binderbauer, T. Tajima, et al., "Achieving a long-lived high-beta plasma state by energetic beam injection", *Nature Communications* 6:6897 (2015)
- [63] D. C. Barnes, "Stability of long field-reversed configurations", *Phys. Plasmas* **9**, (2002) 560
- [64] N. Iwasawa, A. Ishida, L. C. Steinhauer, "Tilt mode stability scaling in field-reversed configurations with finite Lamour radius effect", *Phys of Plasmas*, **7** (2000) 931
- [65] S. V. Ryzhkov "Alternative fusion reactors as future commercial power plants", *J. Plasma Fusion Res. SERIES*, **8** (2009) 35
- [66] H. Momota, Y. Tomita, A. Ishida, et.al., "D-³He fueled FRC reactor ARTEMIS-L", *14th*

International Conference on Plasma Physics and Controlled Nuclear Fusion Research, IAEA
CN-56/G-1-1-3, Germany 1992

- [67] A. L. Hoffman and J. T. Slough, "Flux, energy, and particle lifetime measurements for well formed field reversed configurations", *Nucl. Fusion* **26** (1986) 1693
- [68] S. M. Kaye, F. M. Levinton, D. Stutman, et al., "Confinement and local transport in the National Spherical Torus Experiment (NSTX)", *Nucl. Fusion* **47** (2007) 499
- [69] R.J. Goldston, "Energy confinement scaling in tokamaks: Some implications of recent experiments with ohmic and Strong auxiliary heating", *Plasma Physics and Controlled Fusion*, **17** (1984) 87
- [70] T. Roche, S. Dettrick, A. Fontanilla, et al., "Generation of field-reversed configurations via neutral beam injection", *Nature Communications* (2025) 16:3487
- [71] NASA "Follow the NASA Artemis missions" <https://www.nasa.gov/humans-in-space/artemis/>

Publication List of NIFS Series

NIFS-1120 S. Kobayashi, H.Y. Lee, K. Nishioka, Y. Nakamura, S. Nishimura, T. Mizuuchi, S. Kado, K. Nagasaki, H. Okada, T. Minami, S. Yamamoto, S. Ohshima, M. Yokoyama, S. Murakami, K. Watanabe, R. Seki, N. Kenmochi, S. Konoshima, F. Sano
Parallel flow dynamics and comparison with neoclassical transport analysis in NBI plasmas of Heliotron J
Nov. 20, 2014

NIFS-1121 T. Mizuuchi, S. Kobayashi, T. Minami, F. Sano, K. Nagasaki, H. Okada, S. Kado, S. Yamamoto, S. Ohshima, S. Konoshima, L. Zang, H. Y. Lee*, N. Shi†, N. Kenmochi, Y. Ohtani, K. Hada, T. Harada, Z. Hong, M. Kirimoto, S. Kitani, X. Lu, M, Motoshima, K. Murakami, Y. Nakayama, K. Nishikawa, A. Suzuki, S. Tei, M. Yasueda, Y. Nakamura, Y. Nakashima, N. Nishino, G. Motojima, K. Mukai, R. Seki, Y. Suzuki, K. Tanaka, K. Watanabe, M. Yokoyama, S. Murakami
A New Operation Regime for High-Density Plasma in Heliotron J
Nov. 20, 2014

NIFS-1122 T. Imai, T. Kariya, R. Minami, T. Numakura, T. Eguchi, T. Kato. Y. Endo, M. Ichimura, T. Shimozuma, S. Kubo, H. Takahashi, Y. Yoshimura, H. Igami, S. Ito, T. Mutoh, K. Sakamoto, H. Idei, H. Zushi, K. Nagasaki, F. Sano, M. Ono, Y. Mitsunaka
Development of Over 1 MW and Multi-Frequency Gyrotrons for Fusion
Nov. 20, 2014

NIFS-1123 Y. Nakashima, M. Sakamoto, M. Yoshikawa, K. Oki, H. Takeda, K. Ichimura, K. Hosoi, M. Hirata, M. Ichimura, R. Ikezoe, T. Imai, T. Kariya, I. Katanuma, J. Kohagura, R. Minami, T. Numakura, X. Wang, M. Iwamoto, Y. Hosoda, N. Asakura, M. Fukumoto, A. Hatayama, Y. Hirooka, S. Kado, H. Kubo, S. Masuzaki, H. Matsuura, S. Nagata, N. Nishino, N. Ohno, A. Sagara, T. Shikama, M. Shoji, A. Tonegawa, Y. Ueda
Development of Divertor Simulation Research in the GAMMA 10/PDX Tandem Mirror
Nov. 20, 2014

NIFS-1124 T.E. Evans, K. Ida, S. Ohdachi, K. Tanaka, M.W. Shafer, S. Inagaki, M.E. Austin, Y. Suzuki, E.A. Unterberg, the LHD and DIII-D Experiment Groups
Comparative Studies of Static Edge Magnetic Islands in DIII-D and LHD
Nov. 20, 2014

NIFS-1125 M.W. Jakubowski, A. Kirk, W. Suttrop, K. Tanaka, E. Viezzer, E. Wolfrum, A. Dinklage, C. Hidalgo, S. Kaye, G. McKee, S. Mordijck, O. Schmitz and T.S. Pedersen
Influence of Magnetic Perturbations on Particle Transport in magnetic fusion devices
Nov. 20, 2014

NIFS-1126 S. Nishimura
A Laguerre Expansion Method for the Field Particle Portion in the Linearized Coulomb Collision Operator
Sep. 25, 2015

NIFS-1127 T. Nicolas, K. Ichiguchi
Development of a 3D non-linear implicit MHD code
Jun. 13. 2016

NIFS-1128 MITARAI Osamu and YANAGI Nagato
Suppression method of COVID-19 infection by isolation time control based on the SIR model and the analogy of nuclear fusion research
Dec. 01, 2020

NIFS-1129 御手洗 修, 柳 長門
経済とCOVID-19感染症対策を両立させるために ――SIRモデルと制御核融合の数式の類似性を用いて――
To achieve both economic activity and suppression of COVID-19
--Based on the analogy between the SIR model and the controlled fusion--
May 23, 2021 (in Japanese)

NIFS-1130 御手洗 修, 中村一男, 長谷川 真, 恩地 拓巳, 黒田 賢剛, 筒井 広明, 東島 亜紀, 出射 浩, 花田 和明, 増崎 貴
Osamu MITARAI, Kazuo NAKAMURA, Makoto HASEGAWA, Takumi ONCHI, Kengou KURODA, Hiroaki TSUTSUI, Aki HIGASHIJIMA, Hiroshi IDEI, Kazuaki HANADA, and Suguru MASUZAKI
QUEST におけるブラズマ垂直位置制御を考慮した運転シナリオの開発
Development of the operation scenarios based on the vertical plasma position control in QUEST
JAN. 14, 2025 (in Japanese)

NIFS-1131 O. Mitarai, K. Y. Watanabe, Y. Takemura, N. Yanagi, S. Satake, Y. Narushima, and A. Sagara
Comparative studies on the bootstrap current in the standard and inward-shifted LHD configurations in the 1/v and plateau regimes
July 30, 2025

NIFS-1132 O. Mitarai, T. Asai, M. Sakamoto, and Y. Kawamoto
Investigation of D-³He Fusion Ignition in a Field-Reversed Configuration Modeled as a Highly Elongated Spherical Tokamak
April 6, 2026