High Performance Operational Limits of Tokamak and Helical Systems

Kozo Yamazaki\textsuperscript{a} and Mitsuru Kikuchi\textsuperscript{b}

\textsuperscript{a) National Institute for Fusion Science, Toki, Gifu 509-5292, Japan  \\
\textsuperscript{b) Japan Atomic Energy Research Institute, Naka, Ibaraki 311-0193, Japan}
Outline

1. Introduction
2. Achieved operational domain
3. Equilibrium Properties
4. Confinement
5. Stability Limit
6. Density Limit
7. Steady-State Operation
8. Reactor Prospect
9. Summary
INTRODUCTION

For the realization of attractive fusion reactors, plasma operational boundaries should be clarified, and be extended to the higher performance limit.

There are several plasma operational limits:

1. confinement Limit ,
2. stability Limit,
3. density limit, and
4. pulse-length limit.

Here we would like to discuss on a variety of toroidal plasma operational limits focusing on the similarities and differences between TOKAMAK and HELICAL systems.
# Maximum Parameters Achieved

<table>
<thead>
<tr>
<th>Parameter</th>
<th>TOKAMAK</th>
<th>HELICAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Electron Temperature $T_e$ (keV)</td>
<td>25</td>
<td>10 (LHD)</td>
</tr>
<tr>
<td>Ion Temperature $T_i$ (keV)</td>
<td>45 (JT-60U)</td>
<td>5.0 (LHD)</td>
</tr>
<tr>
<td>Confinement time $\tau_E$ (s)</td>
<td>1.2 (JET)</td>
<td>0.36 (LHD)</td>
</tr>
<tr>
<td>$n_i \tau_E T_i$ (m$^{-3}$ s keV)</td>
<td>15x$10^{20}$ (JT-60U)</td>
<td>0.22x$10^{20}$ (LHD)</td>
</tr>
<tr>
<td>Stored Energy $W_p$ (MJ)</td>
<td>17 (JET)</td>
<td>1.0 (LHD)</td>
</tr>
<tr>
<td>$\beta$ (%)</td>
<td>40 (toroidal)</td>
<td>3.0 (average) (LHD,W7-AS)</td>
</tr>
<tr>
<td>Line-Averaged Density $n_e$ (10$^{20}$ m$^{-3}$)</td>
<td>20 (Alcator-C)</td>
<td>3.6 (W7-AS)</td>
</tr>
<tr>
<td>Plasma Duration $\tau_{dur}$</td>
<td>2 min (Tore-Supra)</td>
<td>2 min (LHD)</td>
</tr>
<tr>
<td></td>
<td>3 hr. 10min. (Triam-1M)</td>
<td>1 hour (ATF)</td>
</tr>
</tbody>
</table>
**Operation Regime and Reactor Requirements**

Normalized parameters

\[ \beta = \frac{n k T}{B^2} \sim \frac{n T}{B^2} \]

\[ \rho_* = \rho_s / a \sim \sqrt{T / (a B)} \]

\[ \nu_* = \nu_{ei} a / \nu_{th} \sim n a / (\varepsilon^{5/2} T^2) \]

**TOKAMAK**

- SSTR
- DIII-D high-beta

**HELICAL**

- MHR
- LHD high-beta
- FFHR

*Helicaltokamak*
### Similarities and Differences between Tokamak and Helical Systems

<table>
<thead>
<tr>
<th></th>
<th>STANDARD TOKAMAK</th>
<th>STANDARD HELICAL</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Equilibrium</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Plasma Boundary Shape</td>
<td>2D</td>
<td>3D</td>
</tr>
<tr>
<td>Magnetic Field</td>
<td>Toroidal (m,n)=(1,0)</td>
<td>Toroidal (1,0) + Helical (L,M)+ Bumpy (0,M) Ripples</td>
</tr>
<tr>
<td>Components</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Plasma Currents</td>
<td>External + BS Currents</td>
<td>No net toroidal current or BS Current</td>
</tr>
<tr>
<td>q-profile</td>
<td>Normal or Reversed shear profile</td>
<td>Flat or Reversed shear profile</td>
</tr>
<tr>
<td>Divertor</td>
<td>Poloidal divertor 2D</td>
<td>Helical or island divertor 3D</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Physics Properties</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Magnetic shear</td>
<td>Substantial Shear or Shearless in</td>
<td>Substantial Shear</td>
</tr>
<tr>
<td></td>
<td>the core</td>
<td></td>
</tr>
<tr>
<td>Magnetic Well</td>
<td>Well in whole region</td>
<td>Hill near edge</td>
</tr>
<tr>
<td>Radial Electric Field</td>
<td>driven by toroidal rotation &amp;</td>
<td>driven by non ambipolar loss (Helical Ripple)</td>
</tr>
<tr>
<td></td>
<td>(grad-p)</td>
<td></td>
</tr>
<tr>
<td>Toroidal Viscosity</td>
<td>Small</td>
<td>Large (Helical Ripple)</td>
</tr>
<tr>
<td>(grad-j, grad-p)</td>
<td>(grad-j) driven (grad-p)</td>
<td>(grad-p) dominant</td>
</tr>
<tr>
<td>Island, Ergodicity</td>
<td>near separatrix</td>
<td>Edge Ergodic Layer</td>
</tr>
</tbody>
</table>
Advanced Plasma Shapes

Standard Tokamak
M=0
(n=0)

Core Symmetry

Quasi Axi-Symmetry (QA)
Quasi Poloidal Symmetry (QP)
Quasi Omunigenity (QO)
Quasi Helical Symmetry (QH)

Edge Symmetry
Helical Divertor
M/L=4/1

M=2
M=3

M/L=10/2

M=4
M=5

Larger M

Standard Helical

TJ-II
LHD
W7-X
HSX
Magnetic Shear / Well

**Tokamak:**
Shear is changed by current profile.
Magnetic well.

**Helical:**
A variety of shears by helical coil.
Magnetic hill near edge.

---

Graph:
- **Normal or Reversed Shear**
- **Flat or Reversed Shear**

Axes:
- \( 1/q \)
- \( \rho \)

Legend:
- TJ-II
- JT-60U Normal Shear
- LHD
- W7-AS
- JT-60U Current Hole
Confinement Scaling Laws

Comparing between ITER ELMy-H Database and stellarator Database adding LHD data

\[ \tau_{\text{ELMY}} \propto \tau_B \rho^{-0.83} \beta^{-0.50} \nu^{-0.10} \]

\[ \tau_{\text{ISS95}} \propto \tau_B \rho^{-0.71} \beta^{-0.16} \nu^{-0.04} \]

\[ \tau_{\text{ELMY}}^{\text{EXP}} = 0.0365 R^{1.93} P^{-0.63} \tilde{n}_e^{0.41} B^{0.08} \varepsilon^{0.23} I^{0.97} \]

\[ \tau_{\text{ISS95}}^{\text{EXP}} = 0.08 a^{2.21} R^{0.65} P^{-0.59} \tilde{n}_e^{-0.51} B^{0.80} \tau_{2/3}^{0.40} \]
Radial Electric Field & ITB

**TOKAMAK**
Er shear driven by toroidal rotation and grad-p (JT-60U, Shirai)

**HELICAL**
Positive electric field driven by ripple loss in low density regime predicted by Neo-Classical Theory
NC-ITB (CHS, Minami & Fujisawa)
Island, Ergodicity and Divertor

TOKAMAK
Poloidal Divertor
ITER

HELICAL
Helical Divertor
LHD
Island Divertor
W7-AS, LHD

Remote radiation

Short connection length
Rather clean separatrix

Targets
Baffles
Titanium evaporators
Stability Limits
grad-j dominant or grad-p dominant?

Tokamak

Ideal beta agrees with Ideal MHD

Resistive beta agrees with NTM & TM, RWM theories

Helical

Beta obtained beyond Mercier mode

Global mode is still marginal.

JT-60U

LHD

Normalized current \( I/aB \) (MA/m/T)

\( \beta_N \) vs. \( \rho \)

\( \beta_N = 3.5 \)

\( R_{ax} = 3.6 \)

\( R_{ax} = 3.75 \)

Volume Averaged Beta (%)

\( V'' = 0 \)

Unstable Region

JT-60U

PBX-M

DIII-D

ITER

TFTR

ASDEX

TPR

PBX-M

\( \beta_N \) = 3.5

\( q_{edge}/q_0 = 4 \)

\( q_{edge}/q_0 = 6 \)

\( \epsilon \beta_p \)

H-mode

High-\( \beta_p \) mode

Large \( p_0/\langle p \rangle \)

Medium \( p_0/\langle p \rangle \)

High-\( \beta_N \) mode

Equilibrium limit

Magnetic Axis Position (m)

Experiment

Unstable

Equilibrium limit

Grad-j dominant or grad-p dominant?

Ideal beta agrees with Ideal MHD

Resistive beta agrees with NTM & TM, RWM theories

Beta obtained beyond Mercier mode

Global mode is still marginal.
TOKAMAK
(JT-60U, Takeji) ERATO-J code
Global mode driven by grad-j

HELICAL
(LHD, Nakajima) CAS3D code
Localized mode driven by grad-p

Low-n mode is interchange-like.
High-n mode is ballooning-like.
Effects of Wall

Tokamak

Kink-ballooning modes driven by grad-j & grad-p can be easily stabilized by the fitted wall.

Helical

Mode is localized and there is no strong wall effect on pressure-driven mode, but substantial effects on BS current-driven external mode.

(JT-60SC, Kurita)

Terpsichore

Evaluations of Wall

Kink-ballooning modes driven by grad-j & grad-p can be easily stabilized by the fitted wall.

(JT-60SC, Kurita)

Unstable

Ballooning mode is stable in the area: $\beta_N < 6$.

Stable

Position of conductor: $r_w/a$

Unstable

(b/a>1.2)

Stable
Operational Density Regime

Tokamak

radiation collapse
leading to current disruption

Helical

radiation collapse
slow plasma decay

\[ n_{20_{-GR}} = \frac{I_{MA}}{\pi a_m^2} \]

\[ n_{20_{-LHD}} = 0.25 \sqrt{\frac{P_{MW} B_T}{R_m a_m^2}} \]
Disruption-Free in Helical System?

Tokamak - Helical Hybrid
(Current Carrying Stellarator)

W7-A

JIPP T-II

Tearing mode, even NCTM, can be stabilized in helical system?

LHD

m/n=1/1

W_{ex}/a=0.085

Fujita et al.,
IEEE Transaction on Plasma Science
Steady-state Operation

Tokamak
NB Current Drive in JT-60U RS Elmy H-mode (80% bootstrap current fraction)

Helical
~1keV long pulse operation
In LHD (ICRF 0.8MW)

LH Current Drive in Triam-1M (3hr.10min.)

(Triam-1M, Sakamoto, this conference)
Progress on Reactor Designs

Lower-aspect designs are explored.
## Operational Limits

<table>
<thead>
<tr>
<th></th>
<th>STANDARD TOKAMAK</th>
<th>STANDARD HELICAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Confinement</td>
<td>Gyro-Bohm</td>
<td>Gyro-Bohm (Grobal)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Helical Ripple Effect (Local)</td>
</tr>
<tr>
<td>Beta Limit</td>
<td>Kink-Ballooning Mode</td>
<td>Low-n Pressure-Driven Mode</td>
</tr>
<tr>
<td></td>
<td>Resistive Wall Mode</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Neoclassical Tearing Mode</td>
<td></td>
</tr>
<tr>
<td>Density Limit</td>
<td>Radiation &amp; MHD Collapses</td>
<td>Radiation Collapse</td>
</tr>
<tr>
<td>Pulse-Length Limit</td>
<td>Recycling Control</td>
<td>Recycling Control</td>
</tr>
<tr>
<td></td>
<td>Resistive Wall Mode</td>
<td>Resistive mode (?)</td>
</tr>
<tr>
<td></td>
<td>Neoclassical Tearing Mode</td>
<td></td>
</tr>
<tr>
<td>Beyond limit</td>
<td>Thermal collapse</td>
<td>Thermal collapse</td>
</tr>
<tr>
<td></td>
<td>Current disruption</td>
<td></td>
</tr>
</tbody>
</table>
For realization of attractive fusion reactors, better confinement and longer-pulsed operations should be achieved in addition to burning plasma physics clarification.

In tokamak systems, critical issue is to avoid disruption and to demonstrate steady-state operation. In helical systems, high performance discharges should be demonstrated with reliable divertor, and compact design concepts should be developed.

Each magnetic concepts should be developed complementally focusing on above critical issues keeping their own merits, for realization of attractive reactors and for clarification of common toroidal plasma confinement physics.