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High Performance Operational Limits of Tokamak and Helical Systems

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Abstract

The plasma operational boundaries of tokamak and helical systems are surveyed and compared with each other. Global confinement scaling laws are similar and gyro-Bohm like, however, local transport process is different due to sawtooth oscillations in tokamaks and ripple transport loss in helical systems. As for stability limits, achievable tokamak beta is explained by ideal or resistive MHD theories. On the other hand, beta values obtained so far in helical system are beyond ideal Mercier mode limits. Density limits in tokamak are often related to the coupling between radiation collapse and disruptive MHD instabilities, but the slow radiation collapse is dominant in the helical system. The pulse length of both tokamak and helical systems is on the order of hours in small machines, and the longer-pulsed good-confinement plasma operations compatible with radiative divertors are anticipated in both systems in the future.

Keywords:

tokamak, helical system, plasma confinement, stability limit, density limit, steady-state operation

1. Introduction

For realization of attractive fusion reactors, better confinement and longer-pulsed operations should be achieved in addition to ignited plasma demonstration. The burning physics and engineering integration are explored by ITER [1] and wide range of plasma operations will be carried out by using more advanced toroidal systems such as advanced tokamak and helical systems. 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 tokamaks and helical systems. Physics for plasma operational boundaries should be clarified, and be extended to the higher performance limit. A comprehensive comparison has been done by Prof. Wagner [2] by using L-mode tokamak database and medium-sized stellarator database. In this paper, this comparison is updated using Elmy H-mode tokamak database and recent helical confinement database including recent LHD data.

2. Achieved Operational Domain

The maximum plasma parameters obtained in tokamak and helical systems are summarized in Table 1. The highest parameters of tokamak plasmas were obtained in various machines such as JT-60U (highest temperature, highest fusion triple product), JET (longest confinement time and highest stored energy), DIII-D (highest beta), Alcator-C (highest density) and TRIAM-1M (longest duration); on the other hand a number of helical machines is still small and the Large Helical

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Table 1 Maximum plasma parameters achieved in tokamak and helical systems.

	TOKAMAK		HELICAL	
Electron Temperature T_e (keV)	25	(ASDEX-U, JT-60U)	10	(LHD)
Ion Temperature T_i (keV)	45	(JT-60U)	5.0	(LHD)
Confinement time τ_E (s)	1.2 1.1	(JET) (JT-60U, NS)	0.36	(LHD)
Fusion Triple Product $n_i\ \tau_E\ T_i$ ($m^{-3}\cdot s\cdot keV$)	15×10^{20}	(JT-60U)	0.22×10^{20}	(LHD)
Stored Energy W_p (MJ)	17 11	(JET) (JT-60U,NS)	1.0	(LHD)
Beta Value β (%)	40 (toroidal) 12 (toroidal)	(START) (DIII-D)	3.0 (average)	(LHD,W7-AS)
Density n_e ($10^{20}m^{-3}$)	20	(Alcator-C)	3.6	(W7-AS)
Plasma Duration τ_{dur}	2 min 3 hr. 10min.	(Tore-Supra) (Triam-1M)	2 min 1 hour	(LHD) (ATF)

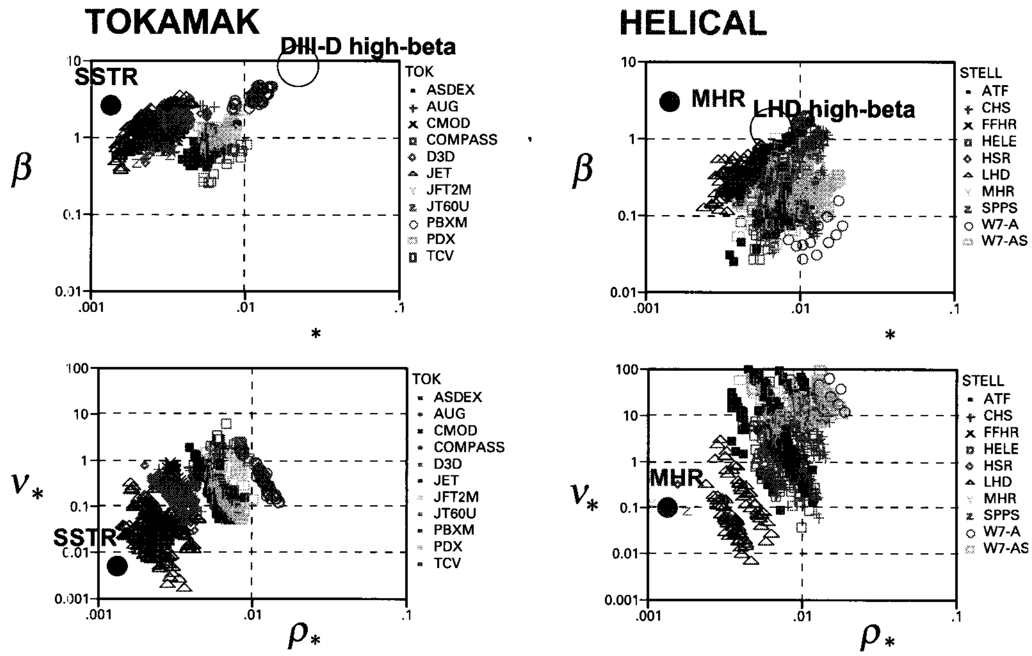

Fig. 1 Operational regimes and reactor requirements in tokamak and helical system.

Device (LHD) produces almost all highest parameters. At present, there is still a parameter gap between tokamaks and helical systems.

Not only absolute parameters, but also normalized parameters are very important for the extrapolation of the present database to the future reactor. Figure 1 shows the operational domain for present machines and tokamak reactor SSTR [3], LHD-type helical reactors MHR [4], by using normalized parameters such as normalized gyro-radius, plasma beta value and collsionality:

$$\rho_* = \rho_s / a \sim \sqrt{T} / (aB),$$

$$\beta = nkT / B^2 \sim nT / B^2,$$

$$\nu_* = \nu_{ei} a / \nu_{th} \sim na / \left(\varepsilon^{5/2} T^2\right).$$

The tokamak data used here are the Elmy H-mode database (IPB-DB3v5) [1] and data from JT-60U advanced tokamak operation [5], and helical data are the medium-sized helical machine database [6] in addition to new LHD data [4]. For extrapolation to the reactor, a few factor reduction in ρ_* is required for tokamaks; on the other hand, the helical system should make access to the one order of lower ρ_* regime in the future. Even in the present helical database the low collisionality regime for reactors has been already explored.

3. Equilibrium Properties

The standard tokamak is characterized by axi-symmetric plasma shaping and external plasma current; the standard helical system is 3-dimensional configuration and net-current-free operation. These different plasma shapings give rise to different magnetic confinement properties. Table 2 shows similarities and differences between tokamak and helical systems.

Example of rotational transform for tokamak and helical systems is shown in Fig. 2. In tokamak systems, magnetic shear is easily modified by the plasma current distribution, for example normal shear discharges and current hole discharges in JT-60U [7]. These shear profiles make strong effect on the production of confinement improvement modes. On the other hand, a variety of magnetic shear configurations can be produced by choosing helical coil systems. The q-profile is reversed or flat, and the magnetic hill region exists near the plasma edge in the conventional helical system.

The divertor configurations strongly depend on the plasma shape. The 2-dimentional tokamak system has poloidal divertor with remote radiation. In helical

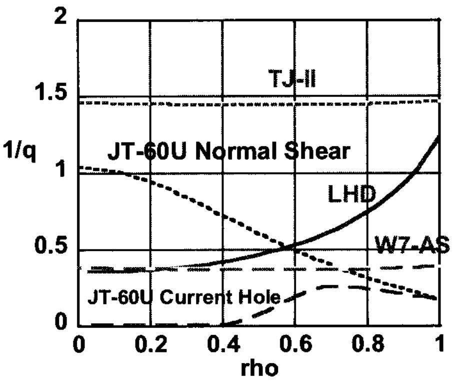

Fig. 2 Magnetic shear for tokamak and helical systems.

Table 2 Similarities and Differences between Tokamak and Helical Systems.

(A) Magnetic Configuration and Equilibrium

	STANDARD TOKAMAK	CONVENTIONAL HELICAL
Plasma Boundary Shape	2D	2D
Magnetic Field Components	Toroidal (m,n) = (1,0)	Toroidal (1,0) + Helical (L,M) + Bumpy (0,M) Ripples
Plasma Currents	External + BS Currents	No net toroidal current or BS Current
q-profile	Normal or Reversed shear profile	Flat or Reversed shear profile
Divertor	Poloidal divertor 2D	Helical or island divertor 3D

(B) Physics Properties

	STANDARD TOKAMAK	CONVENTIONAL HELICAL
Magnetic shear	Substantial Shear or Shearless in the core	Substantial Shear
Magnetic Well	Well in whole region	Hill near edge
Radial Electric Field	driven by toroidal rotation & grad-p	driven by non ambipolar loss (Helical Ripple)
Toroidal Viscosity	Small	Large (Helical Ripple)
grad-j, grad-p	grad-j driven grad-p drive	grad-p dominant
Island, Ergodicity	near separatrix	Edge Ergodic Layer

systems, helical divertor concept with rather large divertor space is adopted in LHD. In the design of modular helical systems the island divertor concept is explored and its effectiveness is demonstrated [8]. The divertor and scrape off layer are related to ergodicity and magnetic island, and differences in stochastic magnetic layers give rise to differences in the performance of plasma confinement.

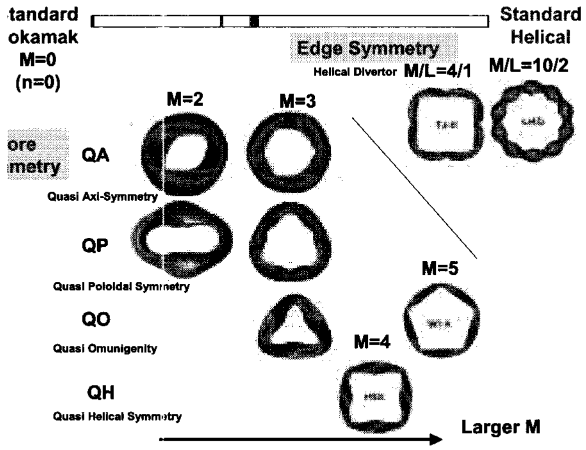

Fig. 3 Advanced 3-dimentional plasma shaping.

These properties shown in Table 2 are described mainly for standard tokamak and conventional helical configurations. At present, various advanced plasma shapings for helical systems are proposed as shown in Fig. 3. Some of them are sorts of tokamak-helical hybrid aiming at disruption-free steady-state operations.

4. Confinement

The global plasma confinement scaling laws in tokamak and helical plasmas are well established, Elmy H-mode IPB98(y) [1] for tokamak and ISS95 for helical systems [6];

$$\tau_{\mathrm{E}}^{\mathrm{ELMY}} = 0.0365 R^{1.93} P^{-0.63} \bar{n}_{\mathrm{e}}^{0.41} B^{0.08} \varepsilon^{0.23} I^{0.97}, \quad (1)$$

$$\tau_{\mathrm{E}}^{\mathrm{ISS95}} = 0.079 a^{2.21} R^{0.65} P^{-0.59} \bar{n}_{\mathrm{e}}^{0.51} B^{0.80} \iota_{2/3}^{0.40}. \quad (2)$$

Where R, a, P, $\bar{n}_{\mathrm{e}}$, B, ε, $\iota_{2/3}$ are major radius (unit: m), minor radius (m), heating power (MW), line-averaged density ($10^{19}/\mathrm{m}^3$), inverse aspect ratio, and rotational transform at $\rho = 2/3$ in helical systems. These two scal-

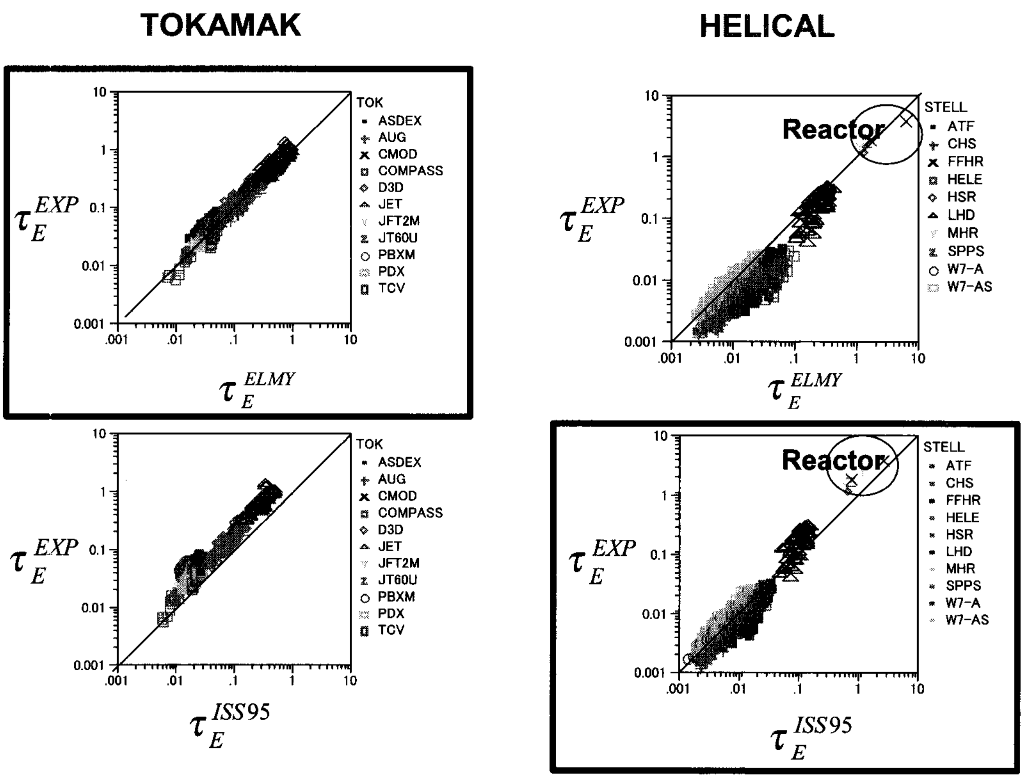

Fig. 4 Confinement scaling laws for tokamak and helical systems.

ing laws are shown in Fig. 4 within solid frames. To compare database of tokamak and helical systems, we used the simple formula of equivalent plasma current I_{equiv} with average minor radius a_{av} for helical systems:

$$q = \frac{aB_{\mathrm{t}}}{RB_{\mathrm{p}}} = 5\frac{\sqrt{\left(1+\kappa^2\right)/2}\,a(m)^2 B_{\mathrm{t}}(T)}{R(m)\,I_{\mathrm{P}}(MA)}$$

$$\equiv \frac{1}{t_{2/3}} = 5\frac{a_{\mathrm{av}}(m)^2 B_{\mathrm{t}}(T)}{R(m)\,I_{\mathrm{equiv}}(MA)}\,. \qquad (3)$$

Figure 4 also shows a comparison between tokamak confinement and helical confinement using $\tau_{\mathrm{E}}^{\mathrm{ELMY}}$ and $\tau_{\mathrm{E}}^{\mathrm{ISS95}}$. The tokamak data scaled by using both confinement laws seem better than the scaled medium-sized helical data, however, the LHD data stays on the ITER Elmy H mode scaling using the above equivalent plasma current. Globally, tokamak and helical transports look similar and of gyro-Bohm type,

$$\tau_{\mathrm{E}}^{\mathrm{ELMY}} \propto \tau_{\mathrm{B}}\,\rho_*^{-0.83}\beta^{-0.50}\nu_*^{-0.10}\,,$$

$$\tau_{\mathrm{E}}^{\mathrm{ISS95}} \propto \tau_{\mathrm{B}}\,\rho_*^{-0.71}\beta^{-0.16}\nu_*^{-0.04}\,,$$

but, local transport seems different. The standard tokamak confinement near the center is determined by sawteeth oscillations, and helical core confinement is affected by helical ripple loss especially in the high temperature and low density regime.

The local transport, especially, the internal transport barrier (ITB) looks different between tokamak and helical systems. The tokamak has clear internal transport barrier on electron and ion thermal transports as obtained in JT-60U experiments. The radial electric field shear is driven by toroidal rotation and pressure gradient. On the other hand, in helical system, radial electric field is driven by ripple loss predicted by neoclassical transport theory. In the low density regime the neo-classical ITB near the plasma center was obtained by the appearance of positive electric field in CHS [9]. The same phenomena have recently been observed in LHD and the detailed physics will be clarified in the future [10].

5. Stability Limit

In tokamak systems ideal beta limits agree with ideal MHD theory, and global beta scaling law is given by

$$\beta_{\mathrm{N}} \equiv \frac{\beta(\%)}{I_{\mathrm{p}}/\left(aB_{\mathrm{t}}\right)} \le 3.5\,. \qquad (4)$$

The pressure peaking effects on plasma stability are also explained by the ideal MHD theory. Moreover, the resistive beta limits agree with neoclassical tearing mode (NTM) or classical tearing mode (CTM) and resistive wall mode (RWM). The kink-ballooning modes, which are current driven mode coupled with pressure driven mode, are restrictive in tokamak discharges. Figure 5 shows the agreement between experimental data and theoretical analysis in JT-60U [11]. On the other hand, in helical system pressure driven modes are dominant. In the LHD experiment, achieved beta value is beyond the Mercier local mode theory, while the global mode analyzed by Terpschore code [12] is still marginal. The unstable mode structure is rather broad in tokamak, on the other hand, the localized mode is crucial in helical system. The low-n mode has an interchange-like structure, and the high-n mode has a ballooning-like one.

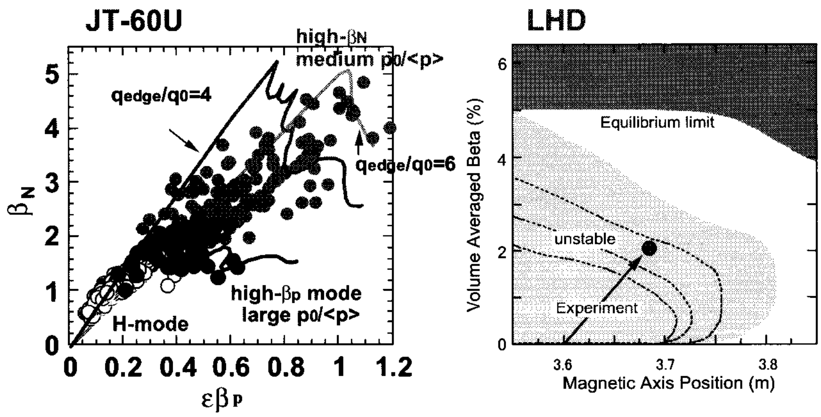

Fig. 5 Stability Limits in tokamak and helical systems.

The current carrying toroidal plasmas are subject to the existence of conducting wall. The global modes are easily stabilized by the fitted wall. In helical systems, the local mode is not linked to the wall, however, the stability of bootstrap (BS) current-carrying helical systems still depends on wall position.

6. Density Limit

The density limit is mainly related to thermal power balance and the radiation collapse. In tokamaks the plasma disruption is often related to the density limits. The operational density regime is plotted by using tokamak scaling (Greenwald scaling [13]) and helical scaling (new scaling with modified coefficient from the helical scaling [14]).

$$n_{20\_\mathrm{GR}} = \frac{I_{\mathrm{MA}}}{\pi a_{\mathrm{m}}^{2}}, \tag{5}$$

$$n_{20\_\mathrm{hel}} = 2 \cdot Min\left[\sqrt{\frac{P_{\mathrm{MW}} B_{\mathrm{T}}}{R_{\mathrm{m}} a_{\mathrm{m}}^{2}}},\ 0.35\frac{P_{MW}}{R_{\mathrm{m}} a_{\mathrm{m}}}\sqrt{B_{\mathrm{T}}}\right]. \tag{6}$$

Figure 6 denotes the density domain of the transport database used in Figs. 1 and 4, not real density limits. This helical scaling can roughly fit to tokamak data as shown in this figure. The density limit of helical plasmas does not seem to be related to the magnetic rotational transform, which is different from the tokamak density limits. The radiation collapse in tokamaks often gives rise to plasma current quench; the helical high-density collapse leads to slow plasma decay.

To produce disruption-free tokamak discharges, one of possible methods is to add external helical field to tokamak plasmas. The complete suppression of major disruption by applying external rotational transform $ι \geq 0.14$ had been demonstrated in W7A [15] and JIPP T-II stellarator [16] twenty years ago. We should check experimentally whether this method is effective even in the BS driven tokamaks.

7. Steady-State Operation

The longest pulsed operation is demonstrated in the TRIAM-1M tokamak for 3 hours and 10minuts [17]. The long-pulsed higher performance discharges are carried out in Tore-Supra. The reactor requirement in steady-state tokamaks is to utilize BS currents and to reduce the circulation power of the reactor plant. The full non-inductive operation with 80 % BS current fractions and 20 % beam driven current has been demonstrated in JT-60U [5].

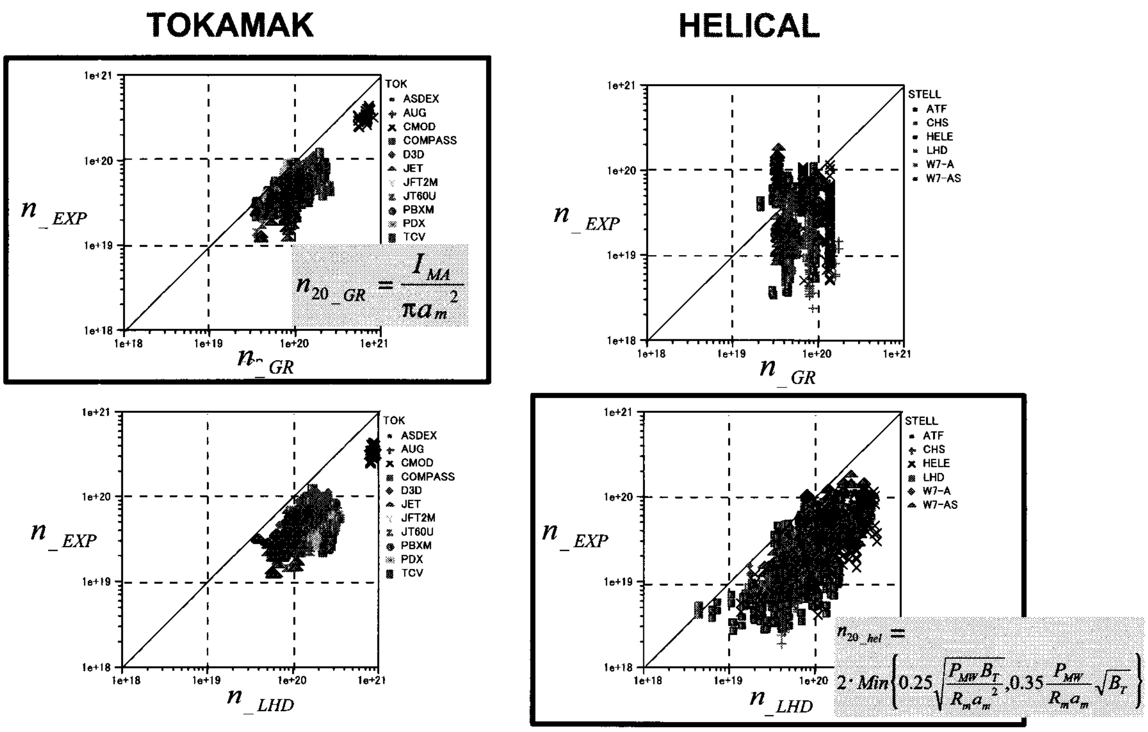

Fig. 6 Operational density regime of tokamak and helical systems.

In helical system, it is easy to keep its magnetic configuration in steady state, and the remained issue is to check compatibility between divertor and plasma confinement.

8. Reactor Prospect

As for reactor designs, one of critical issues for both tokamak and helical systems is compatibility between system compactness and remote maintenance scheme. Especially, helical systems are supposed to be rather large and not attractive from economical viewpoint. Figure 7 shows progress on reactor design for making compact systems. Previous helical reactor design has major radius of ~20 meter, and now low-aspect-ratio designs with major radius of less than 10 meters are explored for the realization of compact and economical reactors based on quasi-axisymmetric (QA) or quasi-poloidal (QP) configurations [18]. A lot of common reactor engineering issues will be sheared between tokamak and helical designs.

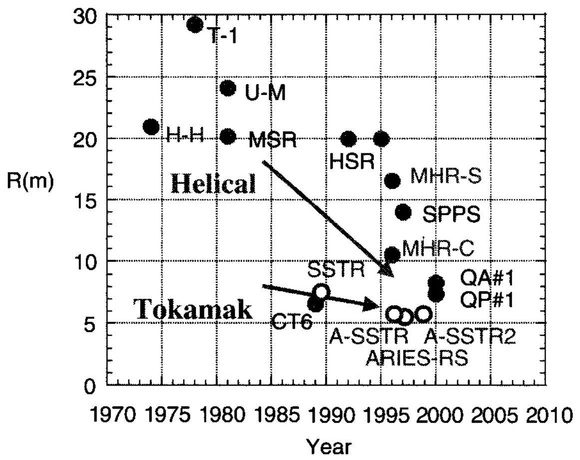

Fig. 7 Progress on reactor design for making compact and economical systems.

9. Summary

Finally we can summarize the operational limits of tokamak and helical plasmas in Table 3. The magnetic configuration in tokamak system can be easily changed by modifying plasma current distribution; in helical systems various plasma shapings by adopting the helical coil system give rise to a variety of magnetic properties. Both global confinement properties are same such as gyro-Bohm scaling. However, local transport is not similar between tokamak and helical system, especially radial electric field formation and internal transport barrier (ITB) properties. The plasma stability of tokamak might be determined by MHD theory related to current driven and pressure driven modes; in helical system the pressure-driven mode is dominant and the achieved pressure gradient is beyond Mercier mode limits.

The realization of attractive fusion reactors, better confinement and longer-pulsed operations should be achieved, in addition to burning plasma physics clarification that will be performed in ITER [1]. 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 explored. Each magnetic confinement concepts should be developed complementally focusing on above critical issues keeping their own merits, for realization of attractive

Table 3 Operational limits in tokamak and helical systems.

	STANDARD TOKAMAK	CONVENTIONAL HELICAL
Confinement	Gyro-Bohm	Gyro-Bohm (Global) Helical Ripple Effect (Local)
Beta Limit	Kink-Ballooning Mode Resistive Wall Mode Neoclassical Tearing Mode	Low-n Pressure-Driven Mode
Density Limit	Radiation & MHD Collapses	Radiation Collapse
Pulse-Length Limit	Recycling Control Resistive Wall Mode Neoclassical Tearing Mode	Recycling Control Resistive mode (?)
Beyond limit	Thermal collapse Current quench	Thermal collapse

reactors and for clarification of common toroidal plasma confinement physics.

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