ASSESSMENT OF TOKAMAK PLASMA OPERATION MODES AS FUSION POWER PLANTS: THE STARLITE STUDY

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Abstract

During the past several years, the ARIES/Starlite Team has investigated the feasibility and potential features of tokamak fusion power plants. The research also has aimed at identifying both the trade-offs that lead to the optimal regime of operation for a tokamak power plant and the critical plasma physics and technology issues. During the initial phase of the Starlite study, an assessment of various tokamak plasma operation modes as candidates for fusion power plants was made. Five different regimes of operation were considered: (1) steady-state operation in the first-stability regime, *e.g.,* ARIES-I, (2) pulsed-plasma tokamak operation, *e.g.,* Pulsar, (3) steady-state operation in the second-stability regime, *e.g.,* ARIES-II and ARIES-IV, (4) steady-state operation with reversed-shear profile, and (5) low-aspect ratio tokamaks (spherical tokamaks). The extent of the plasma physics data base as well as performance as a power plant were considered. Based on the above analysis, the Starlite project has chosen reversed-shear operation for its reference design. This paper summarizes the results of this assessment,

1. INTRODUCTION

The demand for substantial increases in electrical power requirements has been forecast for decades. Fusion offers the promise of an abundant energy source that is not tied to national resources and will have only a minimal impact on the environment. However, fusion will be a new technology in the energy marketplace, it must demonstrate clear advantages over other energy technologies at the time in order to offset the inherent technical risk of a new technology or it will never be widely endorsed. During the past several years the ARIES Team has investigated the feasibility and potential features of tokamak fusion power plants [1–4]. Building upon this research, the Starlite project was initiated to develop the goals and requirements of fusion demonstration and commercial power plants and to assess the potential of tokamaks to meet these goals and objectives. In addition, the research has

Table 1:
TOP-LEVEL REQUIREMENTS AND GOALS FOR COMMERCIAL AND DEMONSTRATION FUSION POWER PLANTS

Element	Demo	Commercial
Must use technologies to be employed in commercial power plant	Yes	Yes
Net electric output must be greater than	75% commercial	N/A
Cost of electricity (COE) must be competitive (in 1995 mill/kWh)	80 (Goal)	65 (Goal)
	90 (Reqmt)	80 (Reqmt)
No evacuation plan required for any credible accident:		
Total dose at site boundary	$<$ 1 rem	$<$ 1 rem
Generate no rad-waste greater than	Class C	Class C
Must demonstrate public day-to-day activity is not disturbed	Yes	Yes
Must not expose workers to a higher risk than other power plants	Yes	Yes
Must demonstrate robotic maintenance of power core	Yes	Yes
Must demonstrate routine operation with less than		
(x) unscheduled shutdowns per year including disruptions	1	1/10
Demonstrate a closed tritium fuel cycle	Yes	Yes
Must demonstrate operation at partial load conditions at	50%	50%

aimed at identifying both the trade-offs that lead to the optimal regime of operation for a tokamak power plant and the critical plasma physics and technology issues.

Based on interaction and advice from the U.S. electric utilities and industry, a set of criteria for fusion power is derived [5–7]. A similar set of criteria has been developed by the EPRI fusion working group [8]. These criteria and associated top-level requirements and goals (Table 1) can be divided in three general categories: (1) Cost, (2) safety and environmental features, and (3) reliability, maintainability, and availability.

Top-level requirements and goals for cost of electricity (COE) were adopted for the Starlite project based on estimated cost of competitive source of electricity at the time of introduction of fusion in the market place [9]. These requirements and goals for COE also are in line with projections of future power plant costs based on energy forecasting models [10]. Safety and environmental requirements are included to circumvent the difficulties experienced by fission and, to some degree, will be faced by fossil fuels in the future. Fusion should be easy to license by the national and local regulating agencies, and be able to gain public acceptance. Fusion power plants should only generate low-level waste (*i.e.,* waste storage time less than a few hundred years). Realization of the full safety and environmental potential of fusion will also help fusion to achieve a cost advantage over other sources of electricity. Fusion power plants can be designed to achieve these criteria only through the use of low-activation material and care in design. However, these requirements result in stringent constraints on the sub-system choices and design. Lastly, it should be demonstrated that Demo and commercial power plants can achieve the necessary degree of reliability. The conceptual design studies can partially address this issue (*e.g.,* by including maintenance considerations in the design). This criterion, to a large degree, should be addressed in the development path of fusion power. Today's experiments are, by their charter, and are not intended to provide detailed engineering data to support the design, construction, and operation of a power plant.

During the initial phase of the Starlite study, an assessment of various tokamak plasma operation modes as candidates for fusion power plants was made. Five different regimes of operation were considered: (1) steady-state operation in the first-stability regime, *e.g.,* ARIES-I [1], (2) pulsed-plasma tokamak operation, *e.g.,* Pulsar [4], (3) steady-state operation in the second-stability regime, *e.g.,* ARIES-II and ARIES-IV [3], (4) steady-state operation with reversed-shear profile, and (5) low-aspect ratio tokamaks (spherical tokamaks). The extent of the plasma physics data base as well as performance as a power plant were considered. In parallel, several options for engineering design (*e.g.,* choice of structural material, coolant, breeder for first wall and blanket) have been developed and assessed. In each area, this assessment was aimed at investigating (1) the potential to satisfy the requirements and goals, and (2) the feasibility *e.g., critical issues* and credibility (*e.g.,* degree extrapolation required from present data base).

In the next section, a brief summary of each physics regime of operation given (Sections 2 through 6). In order to provide a common basis for comparing the cost of power plants, a similar blanket and shield concept has been used which is described in Section 7. This assessment has led to the choice of the reversed-shear as the tokamak plasma operation regime and a self-cooled lithium design with vanadium alloy blanket and in-vessel structures for detailed design (the ARIES-RS study). Detailed accounts of this research can be found in the Interim Starlite [6] and the ARIES-RS reports [7] and in Refs. 5, 11–17.

2. First-Stability Steady-State Regime

In the late 1980's, operation at high bootstrap current fraction as the approach to steady-state operation was proposed by ARIES [1] and SSTR [18] studies simultaneously and independently. In order to reduce the current-drive power, the plasma current is reduced while the bootstrap fraction is maximized. In the first-stability regime, this can be accomplished by operating with a moderately high plasma aspect ratio ($A \equiv 1/\epsilon \sim 4.5$) and low plasma current ($I_p \sim 10$ MA) at a relatively high poloidal beta ($\epsilon\beta_p \sim 0.6$). Detailed MHD and current drive analysis have showed that the maximum bootstrap fraction in this class is about ~70% with most of the driven current located near the magnetic axis, leading to current-drive powers of about ~100 MW delivered to the plasma.

This mode of operation, however, leads to a low value of plasma $\beta \simeq 2\%$ because the important parameter $\epsilon\beta_p$ (which determines the bootstrap fraction) is related to achievable plasma β through

$$(\epsilon\beta_p)\,(\beta/\epsilon) \leq (\beta_N/20)^2\, S \qquad (1)$$

where $S = (1+\kappa^2)/2$ is the plasma shape factor, κ is the plasma elongation, and β_N is defined by $\beta \leq \beta_N(I_p/B_T a_p)$. For a conventional first stability configuration, optimally shaped current and pressure distributions, and with sufficient triangularity in the cross-sectional shape, this equation must be obeyed with a Troyon coefficient of $\beta_N \simeq 3.5$. If a high f_{BS} is desired for steady state operation (such as ARIES-I), one can reduce the current or raise q_*, resulting in a lower β. In order to optimize the equilibrium and bootstrap current profile, these discharges operate with moderately elevated central q_0. For values of $q_0 \simeq 1.3$, the discharges are stable to kink modes without a conducting wall. Higher plasma β values can be obtained by further increasing q_0 but these discharges would require a conducting wall nearby and plasma rotation to stabilize the kink modes.

There is ample experimental data base for this regime, however, operation in discharges with durations longer than the current diffusion time is needed.

3. First-Stability Pulsed-Plasma Regime

Because the recirculating power for inductive current drive is small, it is argued that pulsed-tokamak power plants avoid the problem of non-inductive current drive: the constraint on β_p is removed and plasma β can be higher. However, inductive current-drive also imposes certain constraints on the plasma operating regime. First, because large and expensive poloidal-field coils are needed to supply to inductive flux, these devices also tend to optimize at moderately high-aspect ratio, low-current, and moderately high bootstrap current fraction. Second, because the loop voltage is constant across the plasma, the current-density (induced and bootstrap) and pressure profiles are set for a given pair of density and temperature profiles. In principle, because the current-density profiles cannot be tailored, both the normalized β value and bootstrap fraction are limited ($\beta_N \sim 3.0$, $f_{BS} \lesssim 40\%$). As a result, the plasma β for an optimum pulsed-plasma operation is only marginally ($\sim$25%) larger than that of a steady-state plasma.

In addition, many critical engineering issues have to be resolved. For example, cyclic fatigue and eddy-currents induced in the structural support of the toroidal-field (TF) coils becomes major issues limiting the maximum toroidal field (*e.g.,* the superconducting-coil technology that yields a maximum field of 16 T in a steady-state power plant leads to a maximum field of about 13 T in a pulsed-plasma system). This decrease in the toroidal-field strength more than offsets the gains in plasma β values for a pulsed device, *i.e.,* even though the plasma β is higher in a pulsed device, the fusion power density is lower. Another key issue is the energy storage as the heat-transport system cannot tolerate rapid changes in the thermal power of the power plant during recharge of the OH transformer. Conventional designs for energy storage lead to large and expensive systems. An innovative, sensible-heat storage system in the outer shield was developed for the Pulsar design [4] which has removed the cost of energy-storage as a barrier to pulsed-tokamak power plant.

As a whole, even assuming same reliability and unit cost for components, pulsed power plants are substantially more costly than steady-state ones. On the other hand, this regime of operation is supported by the main body of tokamak experimental data.

4. Second-Stability Stead-State Regime

As is seen from Eq. (1), the only way to increase the plasma β and still have a large bootstrap current fraction is to increase the value of β_N. It may be possible to violate the Troyon limit if certain conditions are met such as through elevated central safety factor ($q_0 \geq 2$) [19] as in ARIES-II/IV designs, reversed magnetic shear [20, 21], or at very low aspect ratios [22]. In addition to these requirements, stabilization of the external kink mode in these regimes requires a close fitting conducting wall and sufficient plasma rotation [23].

Second-stability operation (*i.e.,* operation at elevated values of central safety factor) allows for higher bootstrap current fraction than first-stability operation. While bootstrap current can become as large as the total plasma current (or larger), the bootstrap current-density profile is different than that needed for equilibrium and stability. As a result, current should be driven both on the axis and at the plasma edge while cancellation of part of the bootstrap current in the middle of the plasma is required. Essentially, the optimum configuration has a bootstrap current fraction of ~90% . Compared to the steady-state, first-stability optimum regime, the current-drive power is somewhat smaller and the plasma β is about twice as large (a conducting wall for stabilization of kink modes is required). The critical physics issues for this regime of operation include startup and access to the second-stable operating point, as well as complexity of the current drive system. In addition, there is very little experimental evidence for this mode of operation.

5. Reversed-Shear Steady-State Regime

Reversed-shear plasma operation combines the best features of steady-state, first- and second-stability modes. There is a much better match between the bootstrap current-density profile and that needed for equilibrium and stability and, therefore, higher values of plasma β can be achieved with moderate current-drive power and less complicated current-drive system. The primary characteristics of a reversed-shear plasma are a hollow current-density profile, a non-monotonic safety-factor (q) profile, and relatively peaked pressure profiles. The hollow current-density profile gives rise to a safety-factor profile, which initially decreases from its value at the plasma center to a minimum value, and then rises to its value at the plasma edge. It is the initial decrease in the q-value away from the magnetic axis that provides the negative magnetic shear, which is responsible for stability to $n \to \infty$ ballooning modes. It also appears that in this regime, the plasma transport is suppressed and the a more peaked pressure profile that is consistent with the high β and high bootstrap-current fraction can be sustained. There is ample theoretical research on this regime and some experimental data base is available. An extensive experimental exploration of this regime of operation is currently on-going.

Analyses has showed that to zeroth order, the Cost of the device is independent of the plasma aspect ratio in the range of $A \sim 3 - 4$ (lower plasma β at the higher A is compensated by higher toroidal-field strength on axis and lower current-drive power). In the Starlite study, an aspect ratio of $A = 4$ was chosen because of engineering considerations. At this aspect ratio and with a plasma current of ~11 MA, the maximum theoretical plasma β is

$\sim$5.5% and the bootstrap-current fraction is $\sim$90%. Non-inductive current drive is required to supplement bootstrap current on axis (*e.g.,* fast wave), at the plasma edge (*e.g.,* lower hybrid), and in mid plasma. While several candidate current-drive option exists for driving the current in mid-plasma (*e.g.,* high-frequency fast wave, mode conversion), the data base for these current-drive schemes is very small.

6. Low-Aspect-Ratio (Spherical) Tokamak

Operation at a low plasma aspect ratio (low-aspect ratio or spherical tokamaks) is another approach to achieving high plasma β and a high bootstrap current fraction. Unfortunately, the low aspect ratio of plasma rules out the use of superconducting toroidal-field coils as there is not enough space for a think shield in the inboard. Therefore, a low-aspect-ratio tokamak requires equilibria with a very high plasma β (in order to minimize the Joule losses in the normal-conducting center-post). In addition, because the plasma current is very large ($\sim$30 to 40 MA) in a low-aspect-ratio tokamak, a very high bootstrap current which is also well-aligned to the equilibrium current profile is essential (to minimize current-drive power). Precise alignment of bootstrap current density with the equilibrium current density is essential. Detailed current-drive analysis has shown that even for a bootstrap-current fraction of 95%, the current drive power can easily exceeds several hundreds of MW because the mismatch is usually at the plasma edge and mid-plasma where current-drive systems are inefficient.

Since the power requirements associated with the edge current drive turned out to be so large, we concentrated on a series of equilibria that require only central current drive. We imposes the constraint that the surface averaged parallel current density be a linear function of the poloidal flux ψ interior to the point where the linear profile is tangent to the bootstrap current profile and equal to the bootstrap current outside this point (Figure 1). This constraint completely specifies the profiles and insures that the first derivatives of equilibrium quantities are continuous everywhere. An extensive scan of MHD equilibria led to an equilibrium with $A = 1.25$, $\beta \simeq 40\%$ and over 99% of the plasma current is self-driven (current-drive power is $\lesssim$10 MW). This equilibrium has an elongation of 3.0, which will certainly necessitate a vertical position control system. It is noted that these cases would require a conducting wall to stabilize the external kink mode.

Using the case noted above, preliminary system studies and engineering analyses were conducted. A major conclusion of the system studies was that, even for these cases with low current-drive power requirements, there still was significant recirculating power (*i.e.,* recirculating power fraction $\sim$0.6) associated with the resistive losses in the copper center post and return legs of the toroidal field coils. Because of the high-recirculating power fraction, the low-aspect ratio tokamak is quite sensitive to choice of plasma β, design of the center-post, and detail analysis of the current drive requirements. Other studies suggest lower values of recirculating power fraction in a low-aspect ratio power plants.

The spherical tokamak offers unique features and is currently pursued vigorously in the physics community. However, the level of analysis has not reached the stage needed to make sound assessment of this concept. Many critical issues has to be resolved such as: the

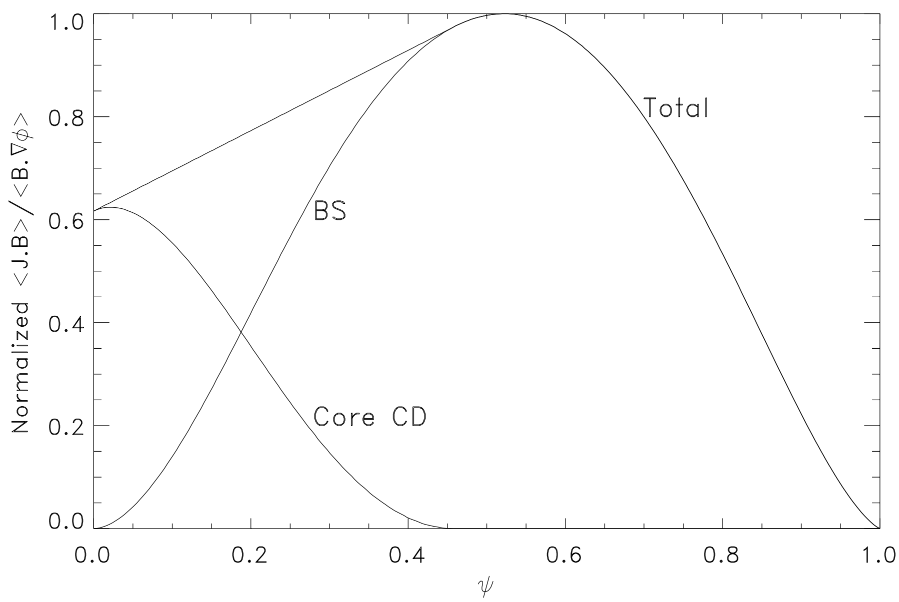

Figure 1: Parallel currents as functions of poloidal flux ψ for a high-bootstrap-fraction spherical tokamak equilibrium which does not require edge current drive.

sensitivity of the current drive power requirements to variations in plasma profiles, center-post design, start-up, design of in-vessel components which are subject to high heat and particle loads, *etc.* The true potential of a low-aspect ratio tokamak power plant can only be assessed after a comprehensive design study.

7. Assessment

A system assessment of the above five tokamak plasma regimes has been performed [6,12]. In order to provide a common basis for comparing the cost of power plants, a similar blanket and shield concept has been used which uses vanadium alloy as structural material of first wall, blanket, and divertor and liquid lithium as the breeder which is a modification of ARIES-II fusion core design. It should be noted that it is not clear if such a design can be utilized in a spherical tokamak (*i.e.,* most probably, the performance of thermal power system would be lower because of the higher wall loading and presence of water as center-post coolant). In addition, we have utilized a maintenance scheme for radial removal sectors (similar to those of ARIES-IV design [3]). This integrated sector arrangement eliminates in-vessel maintenance operations and provides a very sturdy continuous structure able to withstand large loads. The penalty is the increased size of the TF coils needed to allow adequate space for sector removal. Lastly, in order to minimize unscheduled interruption of plant operation, all designs operate at 90% of maximum theoretical β in order to avoid plasma disruption. The major parameters of the five Starlite candidates are shown in Table 2.

In addition to economic performance, an assessment of maturity of each concept was made. To do so, several figures of merits were identified [6,24]. The MHD figures of merits are β/ϵ and $\epsilon\beta_p$ as they indicate progress toward high β discharges with a high bootstrap current

Table 2:
Parameters for the Five Starlite Power Plant Candidates

	FS	PU	RS	SS[(a)]	LAR
Plasma aspect ratio, $A = R/a_p$	4.0	4.0	4.0	4.0	1.25
Major radius, R(m)	7.96	8.68	5.04	6.40	5.00
Plasma minor radius, a_p(m)	1.99	2.17	1.26	1.60	4.00
Plasma elongation, κ_X	1.81	1.80	1.99	2.03	3.40
Plasma triangularity δ_X	0.71	0.50	0.69	0.67	0.55
Cylindrical safety factor q_*	3.77	2.40	2.37	4.60	3.54
Central Safety factor, q_0	1.3	0.7	2.8	2.0	2.9
Stability parameter, $\epsilon\beta_p$	0.54	0.32	0.56	1.22	1.14
Normalized beta, β_N(%)	2.88	2.70	4.76	5.28	6.42
Beta parameter, β/ϵ(%)	8.12	10.00	21.24	12.17	45.36
Confinement ratio, τ^*_{He}/τ_E	10.0	10.0	10.0	10.0	10.0
Ignition parameter, $\beta\tau_E/a_p^2$(%s/m^2)	0.78	2.18	4.17	1.35	11.05
ITER-89P scaling multiplier, H	1.71	2.38	2.40	2.47	3.02
Normalized confinement multiplier, H/q_*	0.45	0.99	1.01	0.54	0.85
Plasma current, I_p(MA)	12.6	15.0	10.3	7.72	40.1
Bootstrap-current fraction, f_{BS}	0.57	0.34	0.89	>1	0.997[(b)]
Current-drive efficiency, γ_B(10^{20}A/W m^2)	0.56	NA	2.02	0.49	34.2
Current-drive power to plasma, P_{CD}(MW)	236.6	0	64.7	199.1	7.3
On-axis toroidal field, B_T(T)	8.99	7.46	7.35	8.37	1.77
Peak field at TF coil, B_{TF}(T)	16.0	13.1	15.7	15.9	14.8
Normalized heat flux, P_{HEAT}/R(MW/m)	71.2	29.5	71.3	89.0	124.2
Recirculating power fraction, $(1/Q_E)$	0.29	0.06	0.13	0.33	0.63
COE (mill/kWh)	99.7	130.2	69.7	92.6	116.0

(a) This design is not optimized to the lowest COE.

(b) Includes diamagnetic current.

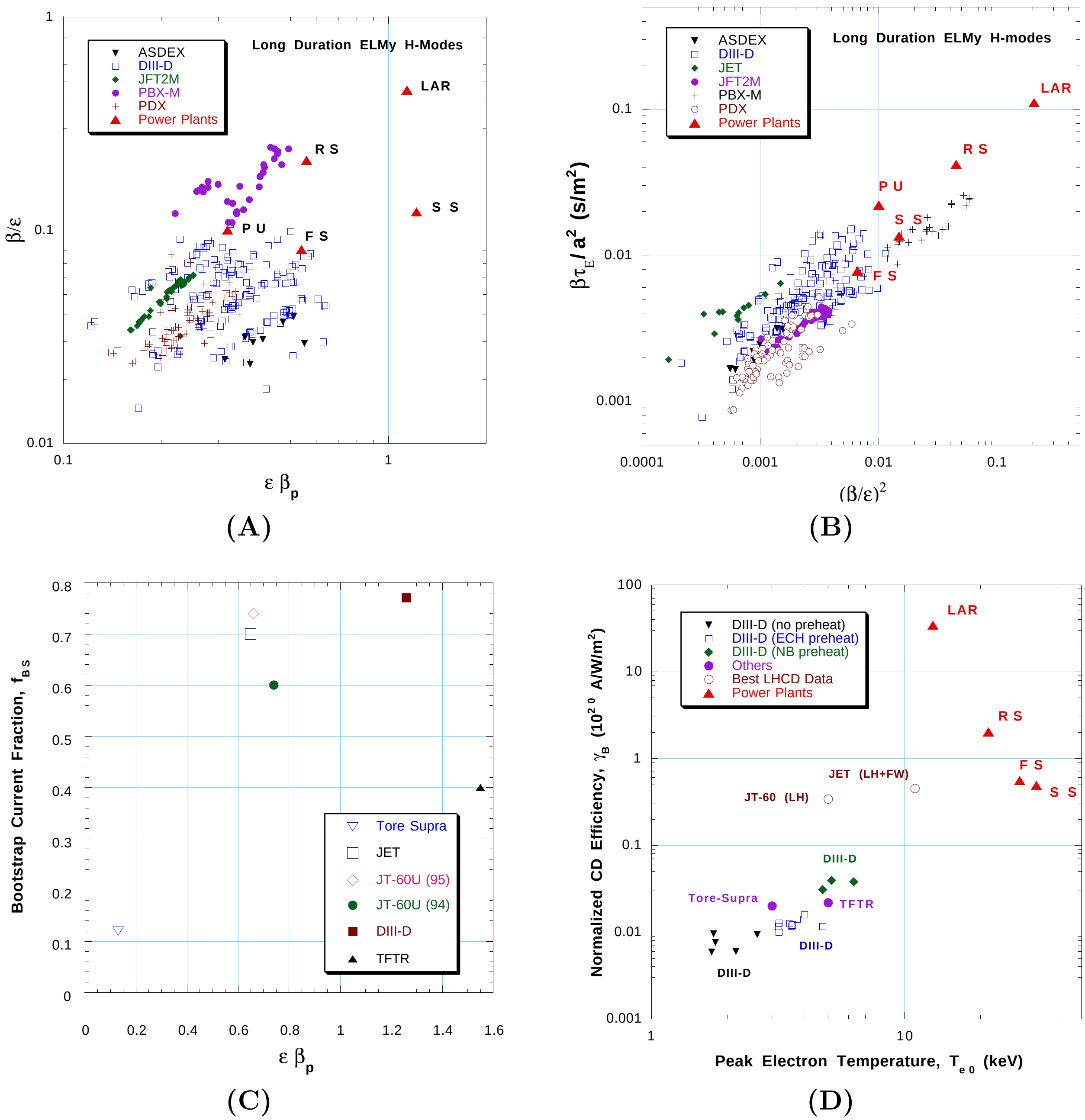

Figure 2: (A) MHD figures of merit for long-pulse ELMy, H-mode discharges in several existing devices; (B) Energy confinement figures of merit for a number existing tokamaks and the Starlite fusion power plants. (Data provided by Stan Kaye, PPPL.) (C) Large tokamak experiments which display substantial bootstrap fraction at high poloidal beta. Note that poloidal beta definitions vary somewhat; and plasma conditions (collisionality, density gradient, impurity content, *etc.*) also vary. (D) Fast-wave current-drive experimental database on DIII-D [25], Tore Supra [26], and TFTR [27]; best data point for lower-hybrid current-drive on JT-60 [28] and for lower-hybrid and FWCD synergy on JET [29]. The five Starlite fusion power plant options (FS, PU, RS, SS, and LAR) are also shown.

fraction. The current-drive figure merit is defined as $\gamma_B = (\overline{n_e}/10^{20}) I_p R / P_{CD}$ which is the conventional measure of current drive efficiency but with the total plasma current appearing to account for bootstrap effects. The energy confinement figures of merit are chosen to be $\beta\tau_E/a^2$ as a function of $(\beta/\epsilon)^2$ to indicate progress towards discharges with high energy confinement time and high plasma β. Figure 2 shows that the existing tokamak data base in two dimensional parameter spaces of these figures of merit. The five Starlite candidates are also shown.

An assessment of the five tokamak physics regimes of operation was made based on the economic performance as summarized in Table 2 and based on the maturity of data base as summarized in Figure 2. The first-stability pulsed-plasma and steady-state regimes are closest to present data base. Of course these regimes should be demonstrated at long-pulse discharges with burning plasmas. On the other hand, the economic performance of pulsed-plasma operation is poor. First-stability steady-state did not achieve the economic requirements for the Starlite project. High-field TF coils can improve attractiveness of this regime of operation. The second-stability regime has a better economic performance but experimental data base for this regime is very small. The data base for the spherical tokamaks is not mature and, in addition, many critical issues remain. Detailed design studies are needed to before the true potential of spherical tokamak power plant can be assessed. The reversed-shear of operation offers the best economic performance. The data base for this regime, while small, is growing rapidly. Based on the superior economic performance and the growing experimental and theoretical data base, the Starlite project has chosen the reversed-shear as the reference plasma operation regime for the ARIES-RS design [7].

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