MFE Concept Integration and Performance Measures
Magnetic Fusion Concept Working Group

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INTRODUCTION

This subgroup of the Magnetic Fusion Concepts Working Group discussed the plans for developing the major magnetic confinement concepts: standard pulsed tokamak, advanced tokamak, spherical torus, compact stellarator, reversed-field pinch, and spheromak The goal was to identify, for each concept, what understanding and capability must be developed to establish its basis for a useful magnetic fusion energy system.

The group initially discussed the concept development process, metrics for development, and the FESAC classification of development levels. This was followed by a discussion of international collaboration opportunities and strategies. Of particular interest was a presentation (by N. Sauthoff) on the National Academy method of (quantifiably) classifying the strengths of a US program by measuring it against the international program.

This was followed by separate consideration of each confinement concept. For each, advocates were asked to present and lead discussions answering the following questions:

1) *For each MFE concept, what are the highest priority new developments required to demonstrate its viability for a practical energy producing system? What facilities and programs are needed to address these? Are there opportunities to minimize costs compatible with a realistic development program?*

2) *What are the perceived strengths and weaknesses of each concept? What opportunities are offered by each concept to reduce fusion development costs and achieve attractive economic and environmental features?*

3) *For each MFE concept, what issues must be resolved in order to motivate and justify advancing to its next stage of development and performance? What are the ideas, plans, and prospects for their resolution, including the entire world program? What metrics should be used to measure progress and readiness to advance?*

4) *What significant roles should the US program seek as part of the international fusion program, and in collaborating with the major international MFE research facilities?*

The answers were summarized during the meeting and discussed further, until there was general agreement by all participants (across the concepts). These summaries formed the basis for the sections which follow, presenting the opportunities, required developments, and metrics for the development of each of the concepts.

I. Pulsed Conventional Tokamak Integration and Performance Measures

1. Benefits

The pulsed conventional tokamak combines magnetic coils and plasma current to magnetically confine a stable toroidal plasma. The plasma current is generated by an electrical transformer so its magnetic field is inherently a pulse of finite duration. Worldwide pulsed tokamak research has demonstrated that fusion energy is feasible, producing up to 16 MW of fusion power (Fig. 1) and 21 MJ of fusion energy in single pulses with a worldwide database nearest to the goal of fusion energy with alpha particle confinement. Able to reliably and controllably produce high temperature plasmas and equipped with extensive diagnostics to benchmark theory and simulation, the pulsed tokamak is an excellent research vehicle for advancing fusion energy science and supporting the development of related magnetic concepts. The pulsed tokamak is technically ready for a high gain burning plasma experiment and enjoys international support for proceeding to this integrated next step to explore the new scientific frontier of burning plasma physics and to develop plasma technologies for power generation.

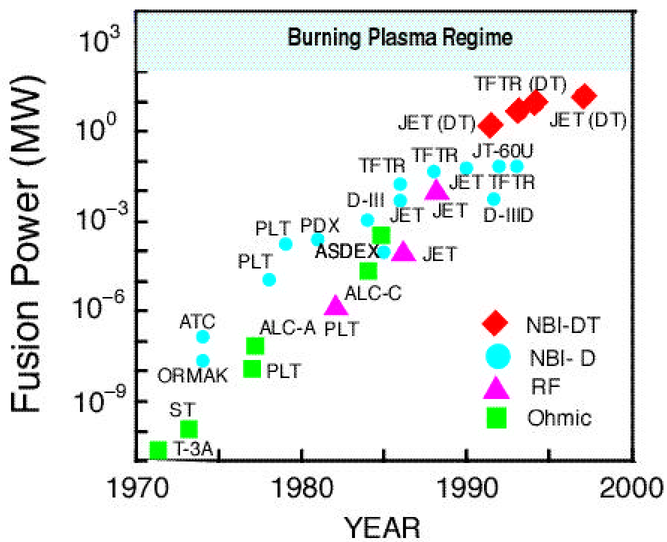

Fig. 1. Tokamaks have made excellent progress in fusion power.

2. Required Developments for Fusion Energy

While being relatively close to fusion energy conditions a number of developments are required for fusion energy. Improvements in the avoidance and mitigation of disruptions at high beta and normalized beta must be developed to increase the reliability of operation and to reduce erosion of plasma facing components. Physics understanding of plasma energy transport, stability, and alpha physics must be developed at sufficient gyroscale with dominant alpha heating. Reliable methods must be demonstrated for handling intense exhaust heat and particle loads with sufficient impurity control. Common to all fusion concepts, engineering materials, breeding blankets, and methods for reliable maintainability must be developed.

3. Other Issues and Concept Weaknesses

The size and cost of the pulsed conventional tokamak power plant leads to a costly development path. Its pulsed nature generates cyclic heat and stress loads and the need for energy storage. Common with the advanced tokamak it must be designed to survive disruption loads and for complex maintenance of superconducting toroidal magnet system and vacuum vessel.

4. Opportunities to Reduce Development Costs

Earlier investments in international tokamak facilities now enables research to be carried out at the performance extension level with only modest upgrades. These facilities (see Table I) are seeking ways to reduce development costs. At the level of fusion energy development (FED) an international consortium is willing to share costs to produce generic science and technology. At an intermediate level a smaller copper burning plasma experiment with lesser technical objectives could be built to decrease the near-term cost and risk, but would delay the eventually needed FED step.

Table I
Characteristics of Operating World Tokamaks

	Plasma Current (MA)	Magnetic Field B(T)	Major Radius R (m)	Comment
Performance Extension Tokamaks				
JET	6.0	4.0	3.0	E.U.
JT–60U	3.0	4.4	3.3	Japan
DIII–D	3.0	2.1	1.7	U.S.
Alcator C–Mod	2.0	9.0	0.65	U.S.
Tore Supra	1.7	4.0	2.3	France (superconducting)
ASDEX Upgrade	1.6	3.1	1.7	Germany
Proof-of-Principle Tokamaks				
FT–U	1.6	8.0	0.93	Italy
TCV	1.2	1.4	0.88	Switzerland
TEXTOR	1.0	3.0	1.75	Germany
JFT–2M	0.5	2.2	1.3	Japan
T–10	0.4	3.0	1.5	Russia
Compass-D	0.4	2.1	0.55	England
Triam-1M	0.15	8.0	0.84	Japan (superconducting)
Concept Exploration Tokamaks (partial list)				
JFT–2M	0.5	2.2	1.3	Japan
ET	0.3	0.25	5.0	U.S./UCLA
Truman-3M	0.18	1.2	0.5	Russia
HBT–EP	0.025	0.35	0.95	U.S./Columbia U.
Steady State Tokamaks (under construction)				
KSTAR	2.0	3.5	1.8	Korea (2004)
HT–7U	1.0	3.5	1.7	China (2004)
SST–1	0.22	3.0	1.1	India (2002)

5. Conventional Tokamak Metrics to Advance to the Next Stage (Fusion Energy Development)

The existing international tokamak physics database and completed technology and development establishes that the pulsed tokamak is technically ready to proceed to a high gain burning plasma experiment at the level of fusion energy development (FED). Deuterium-tritium experiments have already achieved a gain of fusion output power to input power of 0.6. A next stage would require the fusion gain to exceed 5 in order for the alpha particle heating to exceed the auxiliary heating power. Physics and technology options have enabled several next-step burning plasma experiment designs with differing technical objectives (e.g., BPX, ITER, RC–ITER, Ignitor, FIRE, …). Performance metrics to advance to the next step include:

- Adequate MHD stability at $\beta A > 6\%$ ($2.5 < A = R/a < 6$) with scaling of sufficient normalized beta $\beta_N = \beta\ (aB/I) > 2$ and adequate disruption mitigation.
- Adequate energy confinement with a quality factor $H_{89} = \tau_E/\tau_{89} > 1.8$ in regimes of reactor relevance ($T_e = T_i$, $\tau_{He}/\tau_E < 10$, scaling to low ρ_*, at sufficient density $n/n_{GW} \sim 0.7$). Energy confinement projections are shown in Fig. 2.

- Demonstration power handling with P/R > 15 MW/m with adequate core impurity control of Z_{eff} < 1.5.

A summary of further performance metrics are given in Table II and Fig. 3.

6. International Roles to Advance Goals

The U.S. has ceased focusing on conventional tokamak research in favor of advanced tokamak research and no longer has its large TFTR tokamak facility. The U.S. should therefore vigorously collaborate with the large tokamak facilities in Europe and Japan. U.S. experiments pioneered advanced tokamak physics and should aim to sustain an innovative lead by upgrading two national facilities (Alcator C–Mod and DIII–D) for steady-state advanced-tokamak research with current profile control systems. The U.S. has established and should maintain leadership in theory, simulation, diagnostics, and plasma control.

The U.S. should encourage the international parties to construct the redesigned Reduced-Cost International Thermonuclear Experimental Reactor (RC-ITER), maintain a watching brief, and if the parties choose to construct, the U.S. should seek to participate. At the same time, the U.S. should identify contingency smaller next-step burning plasma experiment options, as illustrated in Fig. 4.

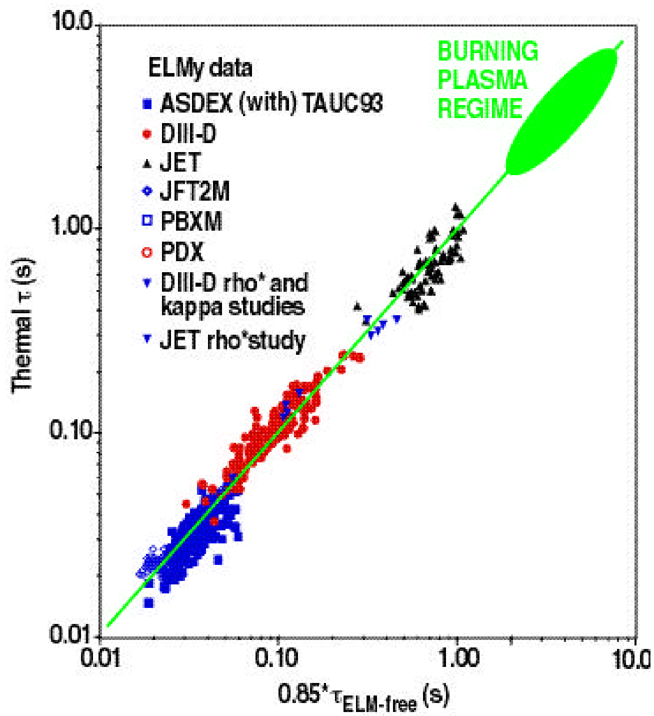

Fig. 2. Tokamak energy confinement studies provide the basis for design of a burning plasma experiment.

Table II
Pulsed Conventional Tokamak Performance Metrics

Attribute	Conventional Tokamak (ITER-EDA)	Best Achieved Values (not simultaneously)	DIII–D Shot 96686	TFTR D–T Shot 80539	JET Shot 47413
MHD Stability					
Plasma pressure relative to magnetic field pressure: $\beta = 2\ \mu_o \langle p\rangle/B^2$ (%)	3	12 (DIII-D)	4.8	1	
Normalized plasma stability factor: β_N = B/(I/aB) (% m-T/MA)	2.3	5 (DIII-D)	3.8	1.8	1.95
Energy Confinement					
Confinement improvement relative to 1989 standard: $H_{89} = \tau_E/\tau_{89L}$	1.8	3.5 (AUG, DIII-D, TFTR)	3.2	2.1	2.3

Heat and Particle Exhaust					
Divertor upstream normalized heat flux q (MW/m^2)	1.0	0.5 (C–Mod)	0.1	–	
Helium ash removal: τ_{He}/τ_E	10	10 (DIII–D, JT–60U, TFTR)			
Integrated Performance					
Fusion power (MW)	1500	16 (JET)	–	10.7	10
P_{fusion}/Pa_{ux}	Ignition (>10)	0.6 (JET)	–	0.27	0.4
Ion/electron temperature: $T_{i/}T_e$ (keV)	30/35	40/15 (TFTR/JET)	10/6	36/13	35/10
Density: n_e (10^{20} m^{-3})	1.0	10 (C–Mod)	0.6	1.0	0.4
Triple product: $n\tau T$ (10^{20} m^{-3} s keV)	10	8 (JET, JT–60U, TFTR)	1	4	7

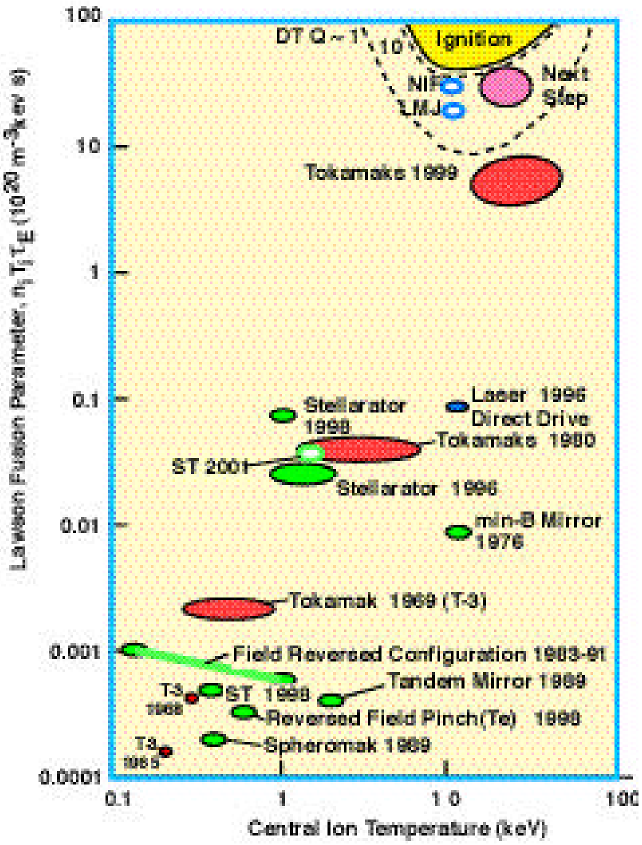

Fig. 3. The conventional tokamak is technically ready for a next-step high-gain burning plasma experiment.

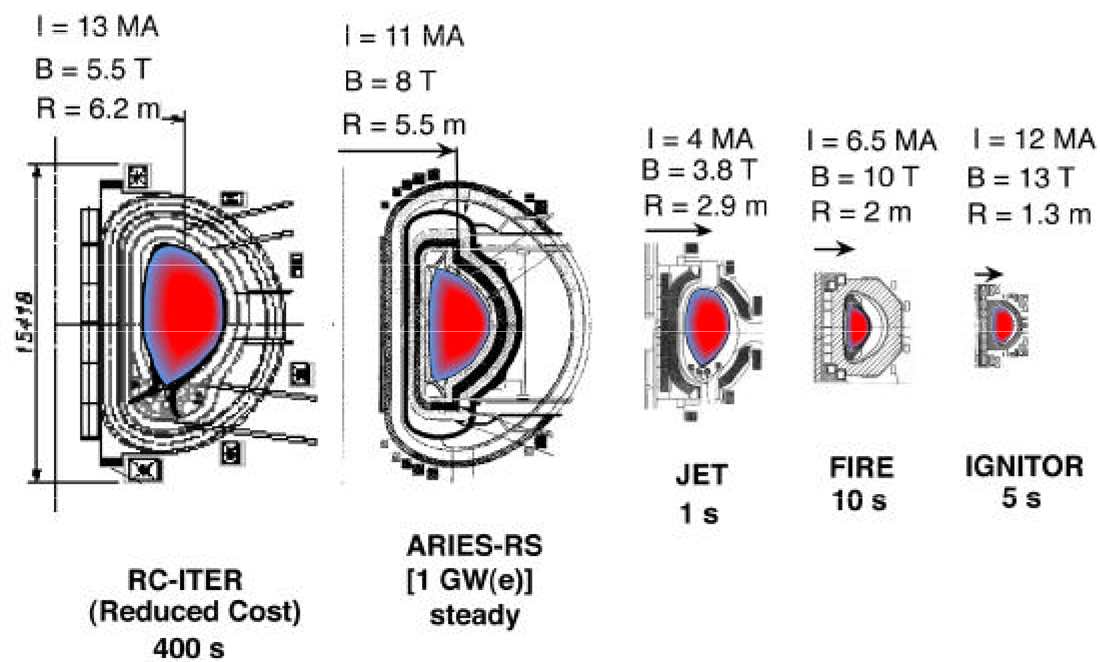

Fig. 4. Potential next-step tokamak burning plasma experiments relative to ARIES-RS power plant.

II. Steady-State Advanced Tokamak Integration and Performance Measures

1. Benefits

The steady-state advanced tokamak has the potential for continuous operation with low recirculating power and thereby avoids cyclic heat and stress loads associated with the pulsed conventional tokamak. Low recirculating power is achieved by utilizing inherent pressure gradient driven bootstrap currents that are maximized by operation at high plasma pressure relative to the poloidal magnetic field pressure (ie high beta-poloidal). In addition, operating at high beta ($\beta = 4\,\mu_o\, nT/B^2$) increases the fusion power since $P_{fusion} \propto \beta^2 B^4 \times$ volume. Reactor studies indicate that the steady-state advanced tokamak leads to attractive reactor prospects with lower size, cost of electricity, and capital cost than a pulsed conventional tokamak (see Fig. 1).

The steady-state tokamak builds on the mature pulsed tokamak and emerging advanced tokamak database at the performance extension level. Examples of advanced tokamak research in high beta plasma stability and in internal transport barrier formation are illustrated in Figs. 2 and 3. Advanced tokamak research facilities with existing extensive diagnostics provide new benchmarks to challenge theory and simulation to advance generic fusion energy science as well as developing plasma technologies for power generation.

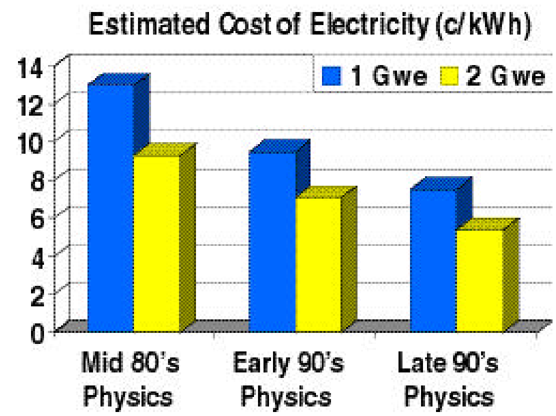

Fig. 1. Steady-state advanced tokamak power plant system studies indicate competitive costs of electricity are attained with advanced physics and technology development.

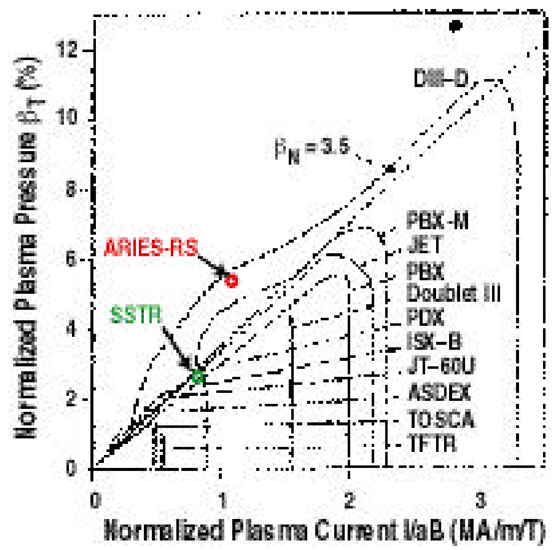

Fig. 2. Accurate guidance on operational boundaries is provided by ideal MHD theory and experiment. Plasma shaping enables increasing I/aB and the b-limit by increasing elongation, triangular shape, and inverse aspect ratio. High values of $b_N = B_T/(I/aB)$, achieved through profile effects and wall stabilization, increase the bootstrap current fraction.

2. Required Developments for Fusion Energy

A number of developments are required to establish an advanced tokamak database that is comprehensive enough to warrant extrapolation. Foremost is achieving simultaneous sustained high plasma beta, good confinement, and high well-aligned bootstrap current fraction. Since advanced tokamak plasmas are somewhat self-organized and operate near stability boundaries, effective disruption avoidance and mitigation are critical. Sustaining such optimized performance will require efficient current drive and effective profile control. An example of the projected stable operating space is shown in Fig. 4. Common to the pulsed tokamak development, a physics understanding of energy transport, plasma stability, and alpha physics must be developed at sufficient gyroscale with dominant alpha heating. The more compact higher performance advanced tokamak will require reliable methods for handling intense exhaust heat and particle loads with adequate impurity control.

Common to all fusion concepts, engineering materials, breeding blankets, and methods for reliable maintainability must be developed.

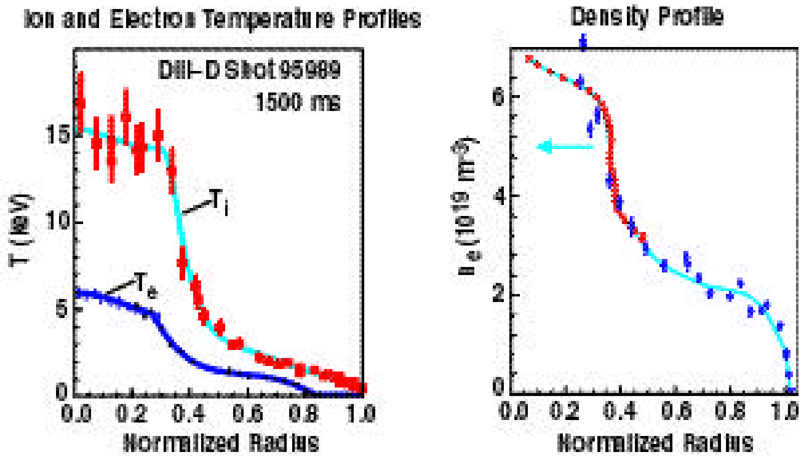

Fig. 3. Localized internal transport barriers improve core confinement. when turbulence is reduced or eliminated by combinations of sheared ExB flow and negative magnetic shear. Control of the steep pressure gradients which can precipate MHD instabilities is a key ongoing research challenge.

3. Other Issues and Concept Weaknesses

Optimization of advanced tokamak performance will require the utilization of current and transport profile control as well as more complex feedback control of MHD modes and equilibrium. Common with the pulsed conventional tokamak it must be designed to survive disruption loads and for complex maintenance of the vacuum vessel and the superconducting toroidal magnet system..

4. Opportunities to Reduce Development Costs

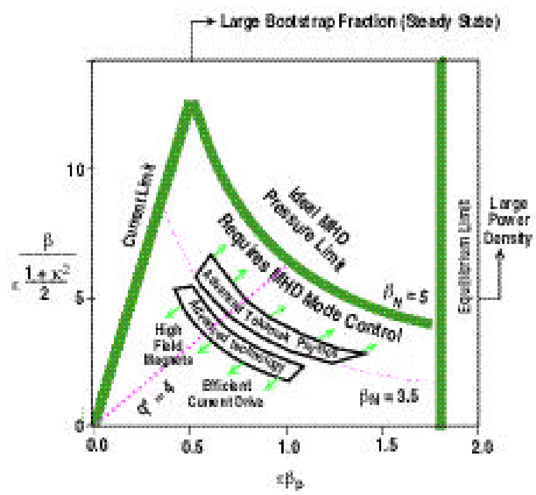

Fig. 4. A compact steady-state advanced tokamak requires operation at high b_N. High power density requires high toroidal beta, b_T. Steady-state requires high I_{bs}/I_p which requires high poloical beta, b_p, high b_T and high b_p require high normalized beta since b_N. $b_T b_p$ µ $(1+k^2/2)$ and b_N ” $b_T/(I/aB)$.

Developing advanced tokamak physics will reduce the cost of subsequent development steps as well as the eventual cost of electricity (Fig. 1). Past investments in existing international tokamak facilities is now enabling advanced tokamak research to be carried out at the performance extension level with modest upgrades for plasma control (e.g., ASDEX–U, Alcator C–Mod, DIII–D, JET, JT–60U). At the level of fusion energy development (FED) an international consortium is willing to share costs to produce generic science and technology (RC-ITER) with some AT capability. At an intermediate level, smaller copper burning plasma experiments (Ignitor, FIRE) with much lower technical objectives (to study transient AT burning plasma physics, but not steady-state physics) could be built to decrease the near-term cost and risk, but would delay the eventually needed FED step. These burning plasma experiments depend on conventional tokamak physics for their baseline design but have varying degrees of capability to develop advanced tokamak physics.

5. Metrics to Advance to the Next Stage (Fusion Energy Development)

The existing international tokamak physics database and completed technology development establishes that the tokamak is technically ready to proceed to a high gain burning plasma experiment at the level of fusion energy development (FED). . Non-stationary advanced tokamak deuterium discharges in JT-60 have achieved a DT equivalent gain of fusion output power to input power of 1.25. The next stage requires the actual fusion gain to exceed 5 in order for the alpha particle heating power to exceed the auxiliary heating power, and for the sustainable time to exceed all relevant time-scales. Performance metrics which would enable a next-step design based on advanced tokamak physics would include:

- Adequate MHD stability at $\beta A > 9\%$ ($2.5 < A = R/a < 6$) with scaling of sufficient normalized beta $\beta_N = \beta\ (aB/I) > 3$ and adequate scalable disruption mitigation.
- Adequate energy confinement with a quality factor $H_{89} = \tau_E/\tau_{89} > 2.2$ in regimes of reactor relevance (T_e T_i, $\tau_{He}/\tau_E < 10$, scaling to low ρ_*, at sufficient density $n/n_{GW} \sim 0.7$.
- Efficient net current drive and profile control: with aligned bootstrap current fraction $f_{BS} > 60\%$ and current drive efficiency $\gamma_B = n_e R I_p / P_{CD}$ projecting to $0.3\infty 10^{20}$ MA/MW•m^2.
- Demonstration power handling with $P/R > 15$ MW/m with adequate core impurity control of $Z_{eff} < 1.5$.

A summary of these steady-state advanced tokamak performance metrics are given in Table I and examples of recent progress are shown in Fig. 5.

Table I
Steady-State Advanced Tokamak Performance Metrics

Attribute	Steady-State Tokamak Reactor (SSTR) Japan	Advanced Tokamak Reactor (ARIES-RS) US	Best Achieved Values (not simultaneously)	DIII–D #96686	JET AT #47413	Metric for AT Next-Step
MHD Stability						
Plasma pressure relative to magnetic field pressure: $\beta = 2 \mu_o \bullet p®/B^2$ (%)	2.5	5	12 (DIII–D)	4.8	1.5	3
Normalized plasma stability factor: β_N = B/(I/aB) (% m-T/MA)	3.2	4.8	5 (DIII–D)	3.8	1.95	3
Energy Confinement						
Confinement improvement relative to 1989 standard: $H_{89} = \tau_E/\tau_{89L}$	1.8	2.4	3.5 (AUG, DIII–D, TFTR)	3.2	2.3	2.2
AT parameter: $\beta_N H_{89}$	5.8	11.5	17 (DIII–D)	12	4.5	6.6
Current Drive						
Plasma duration (s)	Steady-state	Steady state	2 h (TRIAM)	1	2	Steady
Percent bootstrap current (%)	75	89	80 (JT–60U, TFTR)	50		60
Current drive efficiency: $n_{CD}/R/P_{CD}$ (10^{20} A/W•m^2)	–	2	0.4 (JT–60U)	–		
Heat and Particle Exhaust						
Divertor upstream normalized heat flux q (MW/m^2)	2	2	0.5 (C–Mod)	0.1		2
Helium ash removal: τ_{He}/τ_E	10	10	10 (DIII–D, JT–60U, TFTR)			10
Integrated Performance						
Fusion power (MW)	3000	1800	16 (JET)	–	10	200
P_{fusion}/Pa_{ux}	50	29	0.6 (JET)	–	0.4	5
Ion/electron temperature: T_i/T_e (keV)	17/17	21/22	40/15 (TFTR/JET)	10/6	35/12	
Density: n_e (10^{20} m^{-3})	1.4	2.5	10 (C–Mod)	0.6	0.4	
Triple product: $n\tau T$ (10^{21} m^{-3} s keV)	3.3	5.0	5.0 (JET, JT–60U, TFTR)	0.1	0.7	

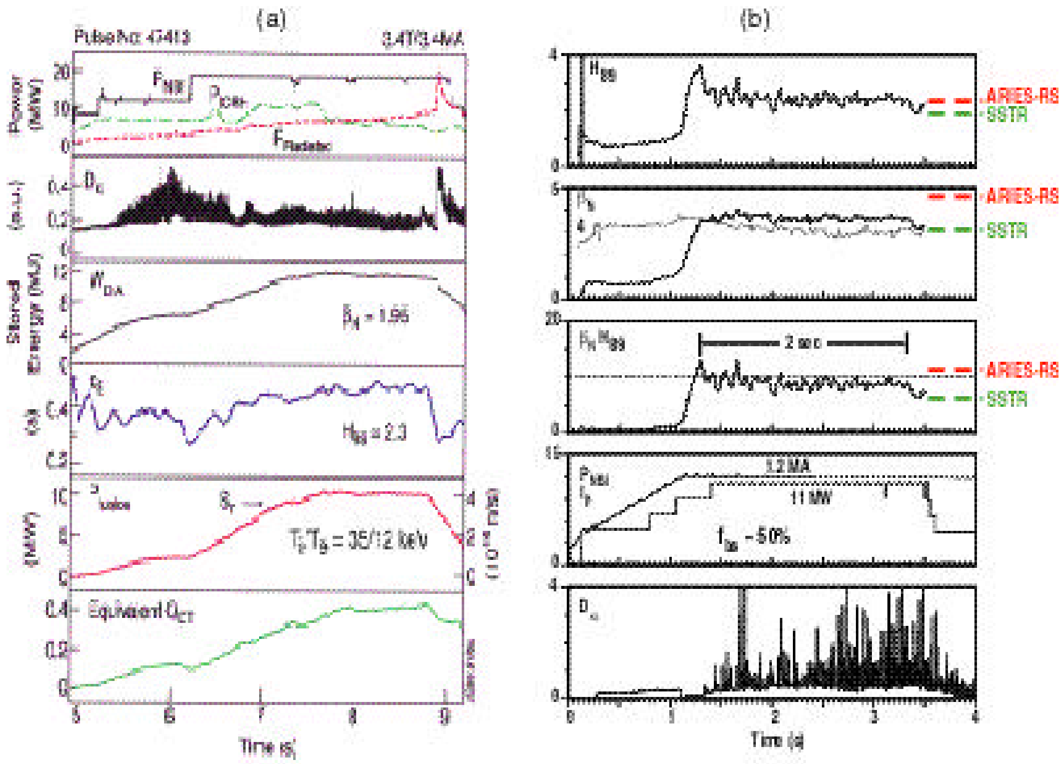

Fig. 5 Two examples of recent progress in advanced tokamak research extending the duration of high performance to 2 seconds..(a) Steady high performance advanced tokamak plasma in JET with high fusion yield, $b_{N,}$ *and confinement having an equivalent* $Q_{DT} = 0.4$*. (b) A steady advanced tokamak plasma in DIII–D with simultaneous high plasma beta and confinement and ~50% bootstrap current fraction.*

6. International Roles to Advance Goals

The US has ceased focusing on conventional tokamak research in favor of advanced tokamak research and no longer has the large tokamak facility TFTR. The U.S. should ocntinue to vigorously collaborate on advanced tokamak research with the large tokamak facilities in Europe and Japan and the other AT facilities (Table II). U.S. experiments have pioneered advanced tokamak physics and should aim to sustain this role in innovation by developing and demonstrating the required profile control systems. . The U.S. has established and should strive to maintain leadership in theory, simulation, diagnostics, and plasma control. The U.S. should collaborate on two future superconducting international steady-state tokamaks (KSTAR, HT-7U) which will be in full AT operation in ~2004.

The U.S. should encourage the international parties to construct the redesigned Reduced-Cost International Thermonuclear Experimental Reactor (RC-ITER) with advanced tokamak research capability, maintain a watching brief, and if the parties choose to construct, the U.S. should seek to participate. At the same time, the U.S. should work to identify contingency smaller next-step advanced tokamak burning plasma options.

Table II
World Advanced Tokamak Research Thrusts

Research Facility	Unique Research Thrust
Performance Extension Tokamaks	
JET (E.U.)	DT capability at large size, LHCD
JT–60U (Japan)	Steady state high performance physics at large size, ECH
DIII–D (GA)	High shape flexibility, high beta, CD divertor, ECH
Alcator C–Mod (MIT)	High field, high density divertor, LHCD
Tore Supra (France)	Long pulse superconducting, LHCD
ASDEX Upgrade (Germany)	AT physics, ECH
Proof-of-Principle Tokamaks	
FT–U (Italy)	High field, IBW
TCV (Switzerland)	High elongation
Concept Exploration Tokamaks	
ET (UCLA)	High beta via omgenity
HBT–EP (Columbia U.)	High beta via feedback

III. Spherical Torus Integration and Performance Measures

1. Benefits

The major strength of the ST is its relatively compact size and simple design which allows for a lower cost facility with easier maintenance and possibly enhanced plasma performance (in terms of β and confinement). The ST is predicted to have an extremely high toroidal beta ($\beta_t \rightarrow 1$) limit, high bootstrap fractions (> 0.7), high I_p/I_{TF}, high normalized current (I_p/aB_{to}), and a reduced disruptivity compared to a tokamak at similar β. Neo-classical tearing modes are predicted to be more stable in the ST configuration than at conventional aspect ratio, and, aided by the large shear flow and geometric effects expected in STs, micro-turbulence is predicted to be reduced or even eliminated.

The results to date are encouraging: $<\beta_t>$ up to 40% (central β of ~ 100%); H-mode confinement; tokamak-like density limits; small and nearly symmetric halo currents (~ 5% I_p) during (sometimes forced) disruptions; and non-inductive startup via CHI up to 200 kA. Calculations exist that indicate that there are MHD stable equilibria with near 100% bootstrap current. These advantages all suggest that a next-step facility could be constructed at much lower cost. Assuming that the present ST experiments are successful, then a PE experiment using an ST would be expected to have lower capital investment than comparable advanced tokamak or stellarator options. Therefore, the ST presents a lower cost development path having options for both electric and non-electiric applications.

2. Required Developments for Fusion Energy

The most critical underlying issues and new developments that must occur for the ST to become a viable energy producing system were discussed. These are listed below with the existing facilities that will begin to address these issues:

- Achieve and understand high β_t stability and confinement at low A (NSTX, MAST, PEGASUS)
- Demonstrate non-inductive current ramp-up and sustainment (NSTX, MAST, CDX-U, PEGASUS, HIT-II, Globus-M, ETE, TST-2, DIII-D, C-MOD)
- Develop self-consistent profiles (current, pressure, heating) with MHD stability, bootstrap alignment, and optimized geometry at high β_t (NSTX, MAST, PEGASUS, DIII-D)
- Understand and avoid energetic particle, resistive wall, edge localized, and neo-classical tearing modes (NSTX, MAST, DIII-D, C-Mod)
- Demonstrate lower disruptivity compared to tokamaks at high β, especially at high-q, high bootstrap fraction (all ST experiments)
- Develop particle and power handling (NSTX, MAST, CDX-U, Globus-M, DIII-D, C-Mod)
- Minimize toroidal field requirements consistent with high β_t and confinement to reduce recirculating power and waste production of future high reactivity facilities (NSTX, MAST, PEGASUS, TS-4)

The present ST program consists of Concept Exploration (CE) and Proof-of-Principle (PoP) experiments located around the globe. A Performance Extension (PE) device will be necessary in the development path, assuming that the PoP and CE experiments are successful.

3. Other Issues and Concept Weaknesses

As a reactor concept, the ST is predicted to have similar Cost of Electricity (CoE) compared to the advanced tokamak (AT) or stellarator. While the capital costs of the reactor plant would be lower due to the ST's simpler construction, the recirculating power and/or sustainment power needed to operate at steady-state reactor conditions are expected to increase the CoE up to the levels of the tokamak or stellarator. The ST has a more demanding power handling requirement due to its compact size. The centerpost also presents a potential damage and activation problem due to the lack of shielding.

4. Opportunities to Reduce Development Costs

The ST offers a path to minimize cost compatible with a realistic development program due to its natural advantage of smaller size and simple design. A detailed example of a development program is outlined in the ST white paper (entitled "Issues and Opportunities for Spherical Torus Research"). The overall cost of developing the ST through the PE stage (< $0.5 billion) and the ensuing Fusion Energy Development (FED) experiment is estimated to be < $2 billion. Thus, developing and demonstrating ST physics will reduce the cost of subsequent fusion development steps.

Development Metrics to Advance to Next Stage (Performance Extension)

In order to advance to the Performance Extension stage, progress must be demonstrated in all of the Required Developments (above). General issues for the PoP to PE step are:

- Demonstrate adequate size, TF, and I_p scaling of confinement and β limits to project to favorable PE performance

- Demonstrate self-consistent profiles (current, pressure, heating) with MHD stability (with and without wall stabilization or feedback control), sufficient confinement, sufficient bootstrap fraction and alignment, and optimized geometry at high β_t

- Develop particle and power control in inner-wall limited (IWL), single-null (SN), or double-null (DN) diverted configurations

Specific metrics have been developed to measure the progress of the present CE and PoP experiments, and success in obtaining these metrics would motivate the next PE-level device:

Metrics	Justification
$\beta_N/\beta_{N,max} > 0.7$	PoP and PE stages only need $\beta_N/\beta_{N,max} \sim 0.5$, while a present reactor design is expected to operate near $\beta_N/\beta_{N,max} \sim 0.8$

$\chi_i/\chi_{neo} \sim 1$	The ST is expected to have minimized turbulence and stability against NTM's, and ion transport should approach neo-classical
$H_f \geq 1$ (ITER-98H)	Scaling of confinement is expected to project to favorable PE performance
Non-inductive ramp-up up to ~ 0.7 I_p	The higher performance ST's will likely not have an ohmic inductive startup capability, so a demonstration of significant non-inductive startup current will be necessary.
$f_{bs} \sim 0.5 – 0.7$	for a reasonable reactor configuration, the bootstrap current should be well-aligned and drive well over 50% of the current
Energetic particles orbits should be confined	Prompt alpha or beam ion loss at levels detrimental to performance is considered unacceptable
Disruption resilience at high-β	A real advantage of an ST over a tokamak could be the reduction of disruptions and disruption severity
Acceptable power loading	The present PoP devices should not scale up to unacceptably high levels of power loading. The PE level devices expect power loadings approaching 10 MW/m^2.

New facilities are not necessary at this time to address these issues, except that a liquid wall experiment at the concept exploration stage should be pursued to address that option of particle and power handling. The existing set of ST machines in the U.S. (NSTX, PEGASUS, HIT-II, CDX-U) and around the world (MAST, Globus-M, ETE, TS-3,TS-4, TST-M, TST-2, HIST) will address these issues in a complementary manner.

5. International Roles

The U.S. program is well connected and coordinated with the international ST program to address the above issues as efficiently as possible. ST research has been an international effort from its inception beginning with the ORNL contribution of a neutral beam injector in support of the START program, a fact in which the community takes great pride. The roles in which the U.S. has assumed leadership or can lead in are the scaling to larger size (NSTX), long pulse and relaxed profiles (NSTX), resistive wall mode stabilization (NSTX), CHI startup (HIT-II, NSTX), HHFW heating and current drive (CDX-U, NSTX, PEGASUS), ultra-low-aspect ratio (A < 1.3, PEGASUS), and high normalized current (PEGASUS).

IV. Low-Aspect-Ratio Stellarator Integration and Performance Measures

1. Benefits

Low-aspect-ratio ("compact") stellarators, those with plasma aspect ratio $A_p = R/\langle a \rangle = 2–4$ (where R and $\langle a \rangle$ are the average major and minor radii of the plasma), have the potential for an attractive reactor because stellarators are intrinsically steady state with no need for external current drive, rotation drive, or feedback control. This leads to low recirculating power in a reactor. The freedom from disruptions observed in stellarators means potentially very low disruption loads. Low-A_p stellarators are predicted to have similar power densities as advanced tokamaks but be stable against kinks, neo-tearing, and vertical instabilities without feedback or a close conducting wall at $\langle \beta \rangle > 5\%$. This combination may result in a more reliable reactor system that is simpler to control. In addition to lower cost for a given fusion power in a reactor, lower A_p means larger plasma size for a given cost for an experiment. The use of the bootstrap current to produce part of the confining poloidal field in low-A_p stellarators and the large degree of external control over the configuration properties allow studies that broaden our understanding of toroidal confinement physics.

Figure 1 illustrates the potential gain for a compact stellarator reactor. The SPPS Stellarator Power Plant Study reactor in Fig. 1 (with R = 14 m and A_p = 8) has many of the physics properties of the R = 22 m HSR reactor with A_p = 12 that is based on Wendelstein 7-X (W7-X), but has an estimated cost of electricity (COE) similar to the R = 5.5 m ARIES-RS advanced tokamak reactor. The low A_p offered by compact stellarators coupled with low recycled power may further reduce the reactor COE.

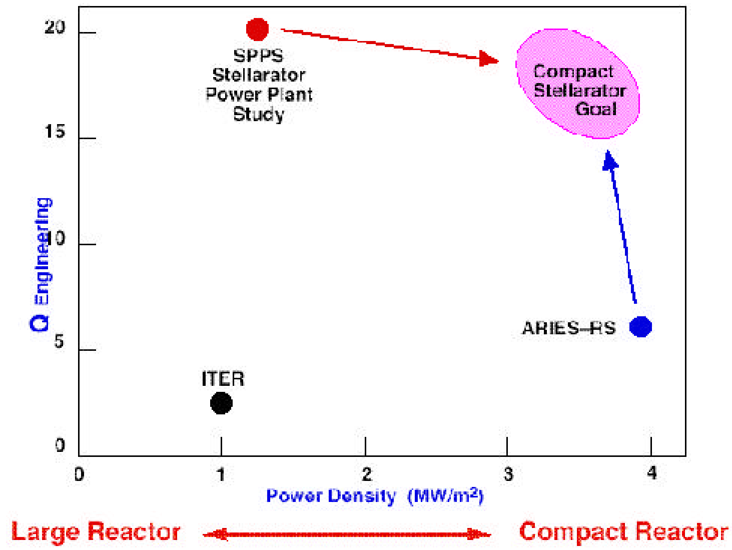

Fig. 1. Potential advantage of a Compact Stellarator reactor.

2. Required Developments for Fusion Energy

The long-term goals for development of compact-stellarators are:

(1) immunity to disruptions at $\langle \beta \rangle > 5\%$ with a self-consistent bootstrap current in steady-state operation and understanding the beta limiting mechanisms

(2) compatibility of the bootstrap current (and its control) with operation at high b and low n*

(3) energy confinement a factor >2 better than the ISS95 stellarator confinement scaling

(4) low neoclassical ripple transport and acceptable loss of alpha particles in a reactor

(5) practical steady-state power and particle handling schemes that are extrapolatable to a reactor-relevant configuration

(6) reactor designs with good plasma-coil spacing and coil utilization.

(7) reactor-relevant plasma parameters (T_i > 10 keV, <b> > 5%, $n\,\tau_E T > 10^{20}$ keV•s•m^{-3})

The large currentless stellarators LHD and W7-X will contribute to the goals (3) – (6) at medium to high aspect ratio, but not for $A_p < 6$. Issues (1), (2), and (7) are the focus of the proposed U.S. proof-of-principle (PoP) compact stellarator program to develop low-A_p stellarators. The capabilities of the world stellarator facilities are indicated in Fig. 2 where the dot sizes are proportional to the plasma cross-sectional area. Here underlining indicates U.S. experiments, black the existing experiments, blue those under construction or modification, and red the proposed compact stellarators.

Although many of the issues listed above can be addressed in the higher-A_p stellarators in Fig. 2, a U.S. proof-of-principle (PoP) compact stellarator program is needed to address issues the for low-A_p configurations. Two new complementary approaches have arisen: quasi-axisymmetry (QA), which has tokamak-like symmetry properties and uses the bootstrap current to produce about half of the poloidal field, and non-symmetric quasi-omnigeneity (QO), which approximately aligns bounce-averaged drift orbits with magnetic surfaces and aims at a small bootstrap current. The proposed program has two new experiments: the QA National Compact Stellarator Experiment (NCSX) PoP facility (with R = 1.4 m, $\langle a \rangle$ = 0.42 m, B = 1.2 T, P = 6-12 MW) and the Quasi-Omnigeneous Stellarator (QOS) concept-exploration-level experiment (with R = 1 m, $\langle a \rangle$ = 0.28 m, B = 1 T, P = 1-3 MW). NCSX would focus on beta limits and disruptivity at high beta and QOS would verify the QO reduction in neoclassical transport and bootstrap currents. They will allow plasma parameters beyond those achievable in the present U.S. stellarators HSX and CAT-U, and would extend stellarator research to much lower aspect ratios. Combined with data from the world stellarator program, these new experiments would produce the physics data base needed to decide whether to proceed to a next step in the compact stellarator line.

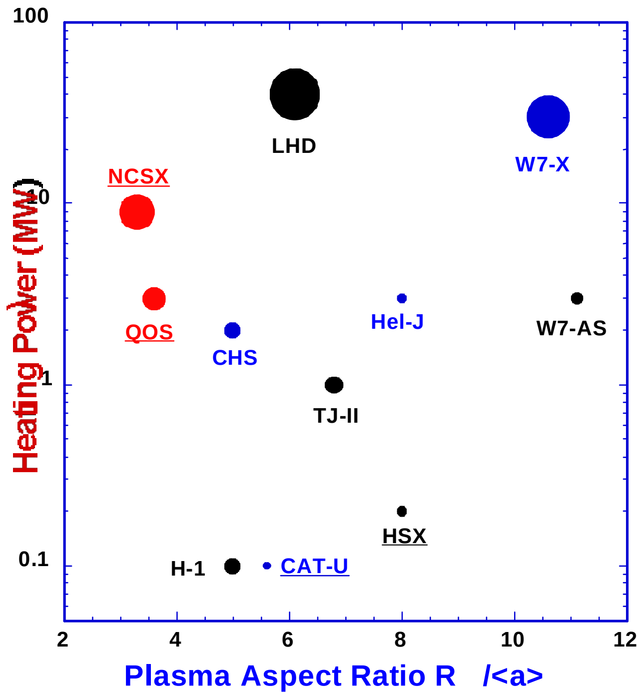

Fig. 2. Comparative sizes and heating powers for the world stellarators.

3. Other Issues and Concept Weaknesses

Low-A_p stellarators use nonplanar coils to create a large part of the confining poloidal field. Nonplanar coils may be more costly to make and have a lower ratio B_0/B_{max} than planar TF coils. Here B_0 and B_{max} are the values of the magnetic field on the axis and the maximum field on the coils. Reducing the cost of nonplanar coils will benefit from decades of experience constructing high-accuracy nonplanar coils for stellarators and more recently the large superconducting nonplanar coils for the Large Helical Device (LHD) and W7-X. One of the goals of compact stellarator reactor studies will be to optimize coils with a higher value for B_0/B_{max}.

4. Opportunities to Reduce Development Costs

Significant reduction of development costs can be achieved by a modest U.S. compact stellarator program leveraging off the large world stellarator program. Experience with large superconducting coils is being obtained in the LHD program and large modular superconducting coils are being built and tested in the W7-X program. Divertors based on magnetic islands are being developed in LHD, W7-AS, and W7-X for steady-state operation at high power. The dependence of beta limits and transport on magnetic configuration properties at higher temperatures and lower collisionalities will be obtained at higher aspect ratio on LHD and W7-X.

Important information can also be obtained at low cost from the higher-A_p smaller stellarators in the U.S. program. The Helically Symmetric Experiment (HSX) experiment will provide the first information on quasi-symmetry by verifying the reduction of neo-classical transport for quasi-symmetric configurations, demonstrating a reduction in the direct loss of deeply trapped particles, and showing that restoration of a direction of symmetry leads to lower viscous damping of the plasma rotation on a flux surface. The Compact Auburn Torsatron is being upgraded (CAT-U) to study disruptions in a stellarator with net plasma current.

The proposed PoP program to test the QA (NCSX) and QO (QOS) optimization approaches for low A_p would reduce costs by reusing existing facilities and components (PBX-M, ATF). A cutaway view of the installed NCSX stellarator core with the PBX neutral beams and TF coils is shown in Fig. 3.

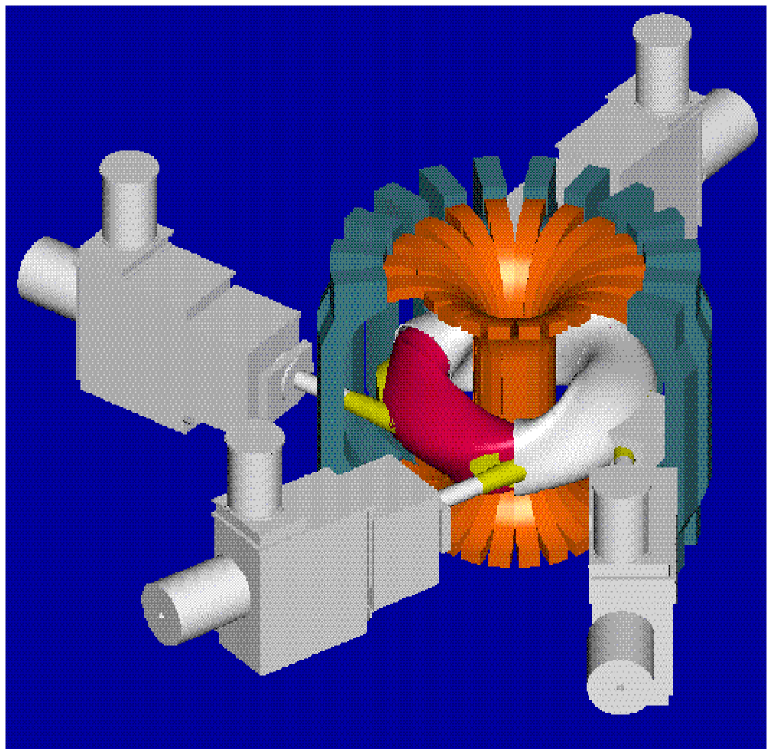

Fig. 3. The NCSX core in the PBX-M TF coils.

5. Metrics to Advance to the Next Stage (Performance Extension/Fusion Energy Development)

The information needed for low-A_p stellarators to advance to the performance-extension or fusion-energy-development stages is summarized in Table 1. The existence of the two large performance-extension stellarators LHD (with $A_p > 6$) and W7-X ($A_p \sim 12$), coupled with the results from the complementary low-A_p NCSX and QOS, may allow low-A_p stellarators to skip the performance-extension stage.

Table 1. Compact Stellarator Needs for Advancing to Next Stage

Demonstration Needed for	Metric	Source
Performance Extension Stage		
disruption free or minimal disruption loads with ~1/2 the rotational transform from the plasma current at low A_p (2-4)	no disruptions at $\langle\beta\rangle > 5\%$	NCSX
understanding beta limits at low A_p	agreement exp. & theory	NCSX
compatibility of the bootstrap current (and its control) with operation at low A_p	agreement exp. & theory	NCSX, QOS
improved confinement; τ_E scaling	$>2\tau_E$ (ISS95)	NCSX
reduced neoclassical ripple transport at low A_p	<< ISS95	NCSX, QOS
practical steady-state power and particle handling scheme	control of edge density, temp.	LHD W7-X
reactor design with good plasma-coil spacing Δ and coil utilization (B_{max}/B_0)	$R/\Delta <$ ~5 $B_{max}/B_0 < 2.5$	ARIES study
Fusion Energy Development Stage		
plasma parameters (T_i, $\langle\beta\rangle$, $n\,\tau_E T$) that allow extrapolation to a burning plasma	10 keV, 5%, $>10^{20}$ keV•s•m^{-3}	LHD W7-X
acceptable loss of alpha particles in a reactor	<~10%	LHD, W7-X
steady-state operation at multi-MW power level	>30 min.	LHD, W7-X

6. International Roles to Advance Goals

Collaboration with the large international stellarator program in selected areas is an important element of the U.S. compact stellarator program because it provides information on stellarator concept improvement that is not otherwise available in the U.S. program. The international stellarator program is already at the *Performance Extension* stage. It features billion-dollar-class facilities now operating in Japan (LHD) and under construction in Germany (W7-X, 2005) that are designed to demonstrate steady-state disruption-free stellarator operation and a level of performance that allows extrapolation to devices capable of burning plasma operation. These large facilities are supplemented by *Proof-of-Principle* experiments in Japan (CHS), Germany (W7-AS), and Spain (TJ-II).

The information on high-A_p currentless stellarators provided by the world program will extend and complement that provided by the proposed U.S. low-A_p stellarator program. Detailed information on magnetic-island based divertors and boundary control from LHD and W7-AS can be applied to further optimization of compact stellarators. Experience with true steady-state operation on LHD and W7-X at high power can benefit both the compact stellarator and advanced tokamak programs. Continued coordination of the international stellarator program and cooperation on development of 3-D MHD, transport, and coil optimization codes also extends U.S. resources in this area.

V. An Integrated Program for RFP Research

The reversed field pinch (RFP) is a toroidal, axisymmetric, magnetically confined plasma. Its distinguishing features are illustrated in Fig. 1. Like most other toroidal configurations, the confining field is composed of toroidal and poloidal components, but in the RFP both components are produced almost entirely by currents in the plasma. The relatively small toroidal component leads to the RFP's potential reactor advantages, as well as its key physics and engineering challenges. These advantages and challenges are described below.

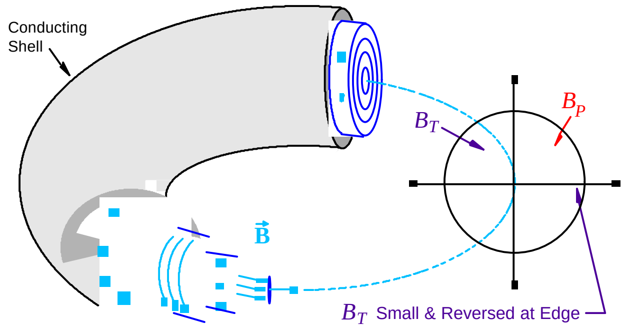

Fig. 1. Magnetic field structure of the RFP.

A proposal to initiate a RFP Proof-of-Principle (PoP) program in the U.S. has occasioned a discussion of the key RFP fusion development issues and the proposed experimental and theoretical program to address them. This proposal has been reviewed by a DOE OFES panel and judged ready to proceed. The status of issues and research plans are covered in greater detail in the PoP proposal document than possible here. For those interested, the proposal, the OFES review panel's report, and the RFP community's response are available in PDF format via the web at http://wwwofe.er.doe.gov/More_HTML/Proof.html. The proposal is included as an appendix in the Summer Study proceedings.

1. Reactor Potential and Cost-Reducing Features

The advantageous features of the RFP concept stem from its relatively small toroidal magnetic field requirement. The RFP's potential as a relatively low cost reactor has been demonstrated in the TITAN reactor study of the late 1980's. Many of TITAN's physics assumptions would be verified by a successful PoP program. The RFP has naturally high beta and high density capability, and is therefore a potentially compact, high power density reactor. Beta values up to ~20% are produced experimentally, but the actual limit is not known (theoretically 50% for ideal interchange stability). As seen in Fig. 1, the magnetic field strength attains its lowest value at the plasma surface where the magnets are located. Consequently the field utilization in the RFP is very high, expressible as high "engineering beta" (plasma pressure normalized to the magnetic field pressure at the coils). The lower field requirement offers possibilities for non-superconducting magnet construction and reduced neutron shielding requirements, allowing single piece maintenance. Since the poloidal field is relatively large, the plasma could be Ohmically heated to ignition. Presently operating RFP experiments with thick, close-fitting conducting shells do not suffer current disruptions. However, past experiments with pulse lengths longer than the shell penetration time allowed the growth of resistive wall modes to large amplitude, terminating the plasma in a disruptive fashion. Control of resistive wall modes is a key issue for the RFP. Also, to date, there is no known constraint on the aspect ratio, leaving it to be determined by engineering optimization.

2. Required Developments for Fusion Energy

The RFP's major physics and engineering challenges are (1) understanding and improving thermal and fast particle confinement using advanced techniques such as current, pressure, and flow profile control, (2) developing efficient steady-state current drive, (3) determining the beta limit, (4) control of resistive wall kink instabilities, (5) MHD configuration optimization (shape, aspect ratio, etc.), and (6) development of compatible power and particle control. The absence of current disruptions (or minimization of disruption loads if they appear) is also necessary for pulses long compared to the resistive-wall time. Although recent progress in improving confinement has been dramatic, understanding and improving confinement remains the highest priority issue. Given the confinement improvement, some view development of efficient current drive as the new highest priority. Clearly both must be established simultaneously. The other issues listed above are more difficult to prioritize and are likely to advance on an opportunity basis. The resistive wall mode problem is a cross-concept issue shared by the advanced tokamak, spherical tokamak, and other high beta configurations. Solutions need to considered in a broader fusion community context, although it is worth noting that 1980's RFP research with resistive shell devices clearly identified resistive wall modes, and in one device (HBTX-1C) a crude but effective helical coil feedback control system demonstrated mode control. This result indicates good promise for mode control in not only the RFP but other configurations as well. An advantage of resistive wall mode investigations in the RFP is that they occur at zero beta. Diagnosing and solving the problem at the concept exploration (CE) level, as in the 1980's RFP program, is therefore cost effective and reliable. The behavior and control solutions for kink modes should not be sensitive to the details of the free energy sources (parallel or perpendicular current gradients).

With the proposed U.S. PoP program in place, the world RFP program (described below) is well poised to address the key issues listed above. The primary focus of the non-U.S. programs is characterizing the confinement of high current RFP plasmas (RFX and TPE-RX) and characterizing resistive wall modes (Extrap-T2). The present focus of the MST program is understanding and improving confinement through transient inductive current drive modifications. The PoP proposal adds to the MST facility essential plasma control tools and key diagnostics (especially profile diagnostics) to improve confinement through refined current profile control, to determine the beta limiting physics, and to develop efficient current sustainment. The control tools to be installed are rf current drive and heating, audio oscillators for Oscillating Field Current Drive, and possibly high power neutral beam heating. In addition to these MST upgrades, one or more new CE experiments could be constructed in the U.S. to address control of resistive wall instabilities, MHD configuration optimization (shape, aspect ratio, etc.), and/or development of RFP-compatible power and particle handling. A necessary part of the PoP program is increased theory and computation in support of RFP physics, as well as new system studies to incorporate recent developments and guide the physics research program. The PoP program draws on the strengths and leadership of U.S. RFP research in recent years and complements the balance of the world's RFP research.

Given space limitations, we describe here in some detail only the confinement and current sustainment issues and plans since they are highest in priority. Complete discussions of these and the other issues can be found in the RFP PoP proposal document referenced above. Two complementary strategies for improving RFP confinement exist. The older strategy is establishing a favorable current scaling. The world RFP confinement database from a dozen or so various sized (but mostly small) devices exhibits an increase of confinement which depends on plasma minor radius and current. This scaling has a constant-beta character. However, it does not adequately describe confinement with current variation in a single device. Establishing a self-consistent and favorable current scaling is a focus of the non-U.S. RFP research programs.

Detailed investigations of the cause of energy transport have confirmed long-held beliefs that magnetic turbulence produces stochasticity in the RFP core, allowing rapid thermal transport by parallel transport. The smaller magnetic field strength in the RFP makes the plasma more susceptible to magnetic turbulence. Detailed nonlinear, resistive MHD theoretical and computational studies over the past 10-15 years has produced excellent understanding of this turbulence and suggested a mitigating approach—current profile control. This forms a new and alternate confinement improvement strategy. Pioneering experiments using inductive current drive techniques on MST have demonstrated a five-fold improvement in energy confinement. Recently, a dramatic increase in the electron temperature has been achieved (doubling of beta), and the temperature profile changes from flat to peaked as shown in Fig. 2. A simultaneous decrease in the core-resonant tearing modes suggests the magnetic stochasticity is reduced. The plans for MST research in the proposed PoP program are to refine and improve current profile control using improved inductive and new rf current drive tools. The physics goal is understanding and eliminating magnetic turbulence.

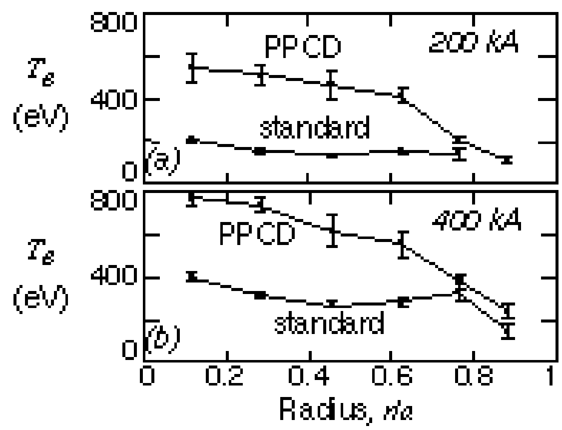

Fig. 2. Electron temperature profiles in (a) 200 kA and (b) 400 kA standard and improved confinement (PPCD) plasmas in MST.

Current sustainment in the RFP is particularly challenging given the large required plasma current and lack of pressure-driven (bootstap) current. One promising candidate has been identified, Oscillating Field Current Drive (OFCD). This is an inductive current drive technique which is accomplished by oscillating at low frequency the toroidal and poloidal loop voltages, but it relies on a current relaxation process to transport edge-driven current into the core. The potential fatal flaw is a possible incompatibility with plasma confinement, since the relaxation process most likely invokes the magnetic turbulence described above. If it can be made compatible with good confinement, it would be a simple steady-state current drive solution with Ohmic efficiency. The electrical resistance in today's larger RFP plasmas is classical to within a factor of two (including a trapped electron fraction correction), so the relaxation process as it presently occurs does not introduce large anomalous resistance. However, if the magnetic turbulence during OFCD is unavoidably large, cooling by energy transport might prevent achievement of reactor plasma temperatures.

Partial current drive by OFCD was tested on the ZT-40M RFP at LANL with positive but inconclusive results. At high current and power, increased plasma-wall interaction (impurity enhanced resistance) was believed to offset the possible current drive. Nevertheless, the theoretical dependence of the relative phase of the loop voltages was confirmed by phasing for anti-current drive, which was successful. Also, probe measurements of the current profile established that the current profile relaxation process was operative throughout the OFCD cycle. A test of OFCD in MST is expected to be conclusive, since the loop voltage requirements are much lower, permitting (in principle) 100% current sustainment by OFCD. Modern wall conditioning using boronization or other techniques should minimize plasma-wall interaction. Low current sustainment or convincing large fraction current drive is relatively easier.

3. Concept Weaknesses and Other Issues

Although the low field requirement for the RFP makes it a potentially compact, simpler reactor concept, it also allows the plasma to be susceptible to magnetic turbulence. As discussed above, reducing this magnetic turbulence is the primary challenge to improving confinement, either through natural current scaling or direct control. The other major weakness is the lack of self-sustained pressure-driven currents. Efficient, steady-state current drive is needed. Existing methods using rf techniques are too costly in recirculating power requirements. If compatible with good confinement, OFCD would be a simple and efficient solution. The possibility for pulsed operation also needs to be investigated for the RFP. The development of liquid metal first walls, a highlight of the Summer Study discussions, could greatly benefit the RFP in providing solutions to the anticipated large wall loading, stabilization of resistive wall kink modes, as well as reducing thermal stress to make pulsed operation practical.

4. The International RFP Program

The present world RFP program is complementary, and the programs at the various facilities are somewhat coordinated through an international IEA working agreement. Three large devices exist: the MST at UW-Madison (a=0.52 m, R=1.5 m), the RFX in Italy (a=0.46 m, R=2.0 m), and the new TPE-RX in Japan (a=0.45 m, R=1.7 m). A smaller, resistive shell device Extrap-T2 operates in Sweden (a=0.18 m, R=1.24 m). The three large devices have similar physical dimensions but different current capability. The RFX device has a 2 MA design, operated 1.2 MA to date. The TPE-RX is designed for 1 MA, and MST operates 0.5 MA. With the PoP proposed additions to MST and appropriately chosen new CE experiments, the international program has the facilities to resolve most of the RFP fusion development issues for advancement to the Performance Extension level of research, as indicated in Fig. 3.

5. Performance Metrics

The RFP community-defined performance metrics to judge the success of the PoP program are shown in Fig. 3. Probably the most difficult projection to the Performance Extension level will be energy confinement. The goal is developing first principles understanding of transport, primarily magnetic transport. Unlike tokamak and stellarator research, which draws on an extensive empirical database formed by a multitude of devices, worldwide RFP research in the next ten years is unlikely to see major new facilities. The present research on confinement scaling with current will most likely not produce a tokamak-like scaling relationship. Instead, RFP research needs to answer fundamentally stated questions: “Can magnetic turbulence be eliminated?” and “If so, what is the underlying transport?” Quantifying the confinement metric in the absence of a well established empirical scaling is less obvious. The stated 10 ms metric is motivated in two ways. First, this is the target confinement for the existing large experiments at 2 MA, stated at the outset of their construction. Second, this is within range of the tokamak empirical database for similarly sized devices operated at the same current (but large toroidal field). The view of the RFP community, however, is not to settle for a particular number, rather strive for physics understanding that allows accurate projected behavior on a fundamental basis.

The beta metric of 15% is motivated by present understanding of MHD physics. It is also a reactor-sufficient beta as determined in the TITAN study. The current drive efficiency metric of 0.1 A/W corresponds to a steady toroidal loop voltage of 10 V, somewhat lower than MST’s standard operation. The absence of large anomalous resistance with OFCD is essential, which ensures Ohmic current drive efficiency. Compatibility with good confinement is explicit and necessary. The success of the PoP program, gauged by these metrics and more importantly the level of physics understanding, will determine the decision to advance to a Performance Extension experiment.

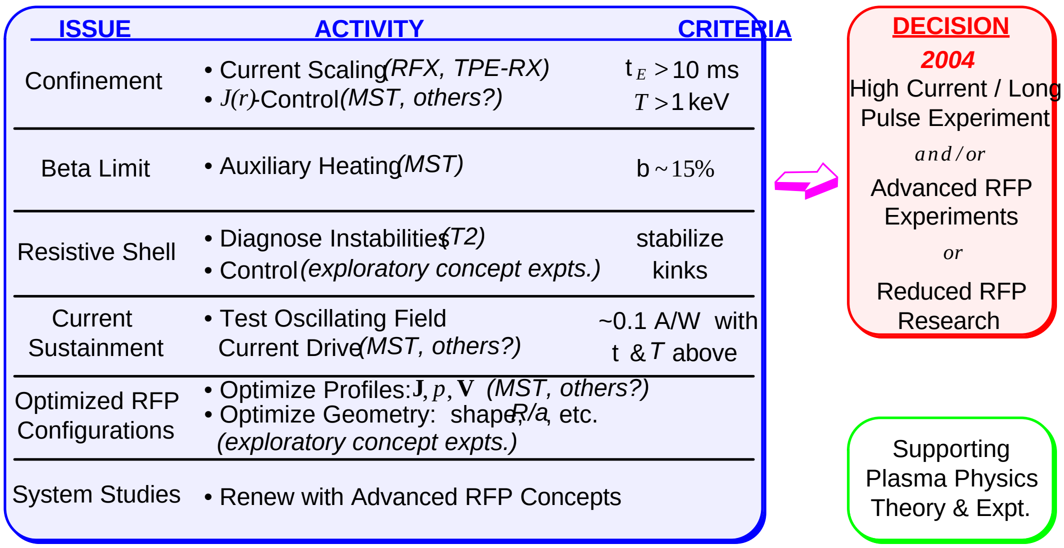

Fig. 3. Metrics and decision options for the RFP PoP program.

VI. Spheromak – Issues and Opportunities

1. Benefits

The spheromak offers three paths to a practical energy producing system: steady-state, pulsed - high beta, and adiabatically compressed (MTF).
A major strength of the spheromak is its simple geometry (no linked coils) and compactness. The simple geometry appears to be especially suited to fluid (liquid) walls because of its lack of a central post and because the relatively high power density in its edge plasma will help shield the core from vapor. Helicity-injection current drive is potentially highly efficient and low cost sustaining a steady-state.

2. Required Developments & Opportunities to Reduce Costs

Energy confinement is the highest priority issue for all spheromaks, and is being addressed in the Sustained Spheromak Physics Experiment. Additional issues are identified in Table 1. Appropriate metrics will be required at each stage of development, and it is anticipated that each stage will require a new experiment and diagnostics to address these issues.

Issue	**CE**	**National PoP Program**		**Enhanced Perform**	**Reactor Exp**
	SSPX*	**PoP**	**Supporting Exp.**		
Energy confinement	**X**	**X**	x	**X**	x
Drive efficiency	**X**	**X**	**X**	**X**	x
Particle control	x	x	x	**X**	x
Global stability & beta limits	x	**X**	**X**	**X**	**X**
Power handling and PWI	x	x	**X**	**X**	**X**
Ignition physics & burn control					**X**

x: will gain information, but not a primary focus for extensive study
X: main subject of experiment – favorable results needed to move to next step.
*Additional CE experiments could also contribute to resolving these issues.

Table 1. Spheromak issues and proposed experimental stage for development

The primary weakness perceived today is the possibly incompatible character of helicity current drive and energy confinement, similar to the RFP. This issue will be addressed in SSPX.

Beta limits and resistive-wall mode stability (for steady-state operation) would be studied in a follow-on Proof of Principle program. The RFP and tokamak programs will also contribute synergistically to understanding these issues.

Steady-state or long-pulse issues such as power handling have been considered only in a preliminary way, but appear manageable. Studies of fluid boundaries and vapor shielding are of particular interest. The potential alleviation of materials neutron damage in reactors would reduce the cost of spheromak development.

The pulsed versions of the spheromak reactor also potentially reduce cost (both development and reactor) by eliminating poloidal field magnets and feedback coils for resistive wall modes. They appear suited to liquid walls. If the physics of confinement, stability, etc., allow one of these pulsed paths, the final development costs should be reduced.

3. Metrics to Advance to the Next Stage (Proof of Principle)

The issues in Table 1 lead to the following metrics for spheromaks to proceed from the Concept Exploration stage to Proof-of-Principle:

- $T_e \sim T_i$ =few hundred eV. Temperatures in this range are both a surrogate for confinement and necessary to conduct physics in a fusion-relevant regime. It is important to achieve $T_e > 100$ eV to ensure that low-Z impurities are burned out. Achieving this will require that high-Z impurities are << 1% of the density, so that $Z_{eff} < 2$-3. $T_e > 200$ eV will demonstrate that the plasma is not effectively mirror confined with fieldline lengths comparable to machine dimensions. Access to the low-collisionallity regime with electron mean-free-paths >> the machine dimensions requires several hundred eV (depending on density). Having the two temperatures comparable is an indication that magnetic activity is not too strong, as $T_i \gg T_e$ is an indication that heating due to resistive tearing or magnetic turbulence dominates electron-ion interactions.

- τ_e scaling favorably with S ($\sim BT^{3/2}$) or other parameters. (S is the Lundquist number.) This scaling is needed to extrapolate to the next experiment and/or reactors. Favorable scaling is predicted by Fowler's energy confinement model, and Rechester-Rosenbluth scaling yields $\tau_e \sim S^{2\alpha}/T_e^{1/2}$; $\alpha > 1/4$ is needed for a reactor. Data is needed to determine the validity of this prediction or to develop improved models.

- Flux and current amplification – e.g. $I_{tor}/I_{inj} = 2 – 3$. In a reactor, these amplifications need to be ~ 100 so that edge effects do not dominate the core. (In a pulsed reactor this condition may be relaxed.)

- Modeling – resistive MHD, transport, etc. must be carried out and benchmarked to experiment. Modeling results are needed to interpret data, and to provide scaling to new experiments. It can also be important for exploring modifications to the spheromak and helicity injector geometry, which are expensive to explore experimentally.

4. International Roles

The US is leading the re-evaluation of the spheromak configuration. At the present time there are no significant spheromak programs outside the US. SSPX has a collaboration to measure ion temperatures with Himeji Institute (Japan), which has done spheromak research in the past. We anticipate that good results from SSPX will stimulate new international research, which will be important to a possible PoP program.